JUN 2 0 1984

MEMORANDUM FOR: G. N. Lauben, Section Leader, Section A, RSB
M. W. Hodges, Section Leader, Section B, RSB
J. E. Rosenthal, Section Leader, Section C, RSB
L. B. Marsh, Section Leader, Section D, RSB

FROM: B. W. Sheron, Chief, Reactor Systems Branch, DSI

SUBJECT: MCGUIRE TECH SPEC ASSIGNMENT

On June 11, 1984, Bob Licciardo of RSB issued his clarification memo documenting his review of the McGuire tech specs. This memo was prepared as part of the post-resolution efforts on this DPO. I have committed to Harold Denton that RSB management will review the issues raised in the document, and will forward the result of the review to SSPB in DL for further action by July 13. Mr. Licciardo's clarification memo is 111 pages long with approximately 5 to 6 items per page. In order to complete the review by July 13, I am dividing the document into 5 equal parts. Each Section Leader will review one part, and I will review one. Enclosed with this memo is the part you are assigned to review.

I believe the best way to approach the review is to try and categorize each concern. As a "first cut," I suggest the following categories be used:

Category A - An acceptable question, typical of the type normally asked during reviews.

Category B - The question seems to challenge the underlying philosophy of the tech specs, rather than a specific question regarding consistency with respect to the safety analyses.

Category C - The question is not clear (you are not sure what the issue is or what is being asked for)

Category D - An unappropriate question. These are questions you may consider are inappropriate for a variety of reasons, such as misunderstanding, beyond the scope of tech specs, etc..

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For each item you review, I would also like you to indicate your perception of its safety significance. I suggest using a (1) for high safety significance, a (2) for medium, and a (3) for low. Keep in mind the following also:

- Some questions can be (and perhaps should be) revised to make them acceptable. If such changes can be quickly made by you, I encourage you to do so.

- Do not hesitate to confer with Bob Licciardo for clarification of individual points.

I suggest you get started on this immediately, and we should plan to get together for a progress report meeting by July 2, 1984.

Original signed by:
Brian W. Sheron

Brian W. Sheron, Chief
Reactor Systems Branch, DSI

Enclosure:
As stated

cc: R. W. Houston

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Lauben

TABLE 3.3-2 REACTOR TRIP INSTRUMENTATION RESPONSE TIMES

Item 1: Manual Reactor Trip

At this time, the licensee proposes that the Response Time (RT) for manual reactor trip is not required by safety analysis. Furthermore, he proposes that in MODES 3 through 5, the only remaining operable trips are those using Source range neutron Flux and they also are not required by Safety Analyses.

Under TABLE 3.3-1, items 2-21 (selected) we have already required the licensee to re-evaluate his position in respect of what neutron Flux trips he intends to propose, together with their related Tech specs to place the reactor in a safe condition in respect to Condition II, III and IV Occurrences in MODES 3 through 5. Until this evaluation and proposal are accepted, the Licensee shall have a Safety Related Manual Trip System to assist in meeting minimum Regulatory Requirements in 10 CFR 50, APP. A. III. Protection and Reactivity Control Systems, and the Licensee shall evaluate and propose as a priority issue. At this time, the proposed T.S is non-conservative in respect to Regulatory Requirements for 10 CFR 50, App. A. III.

Items 5 and 6: Intermediate Range and Source Range Neutron Flux Trips.

As indicated under item Table 3.3-1, items 1-5, these items are proposed as not being protective actions necessary for the FSAR. Analyses already requested will provide a base for determining whether those trips are necessary to protect the plant in MODES 3 through 5. If so, please provide the necessary technical specifications for these response time in conformance with 10 CFR 30.46. If these values are not provided, all related return to reactivity events shall be evaluated by the Licensee with current FSAR requirements for the Safety Analyses Limit of the power range, neutron flux, low setpoint trip which will be required to be OPERABLE.

The current proposals for these trips is non-conservative with respect to other proposals in the T.S; the Licensee shall evaluate and propose.

Item 8: Overpower ΔT.

No response time is provided by the Licensee who proposes that a T.S. on this is Not Applicable.

Please comment on the fact that this reactor trip is proposed in Reference 5 Table 7.2.1-3 (3 of 5) as applying to five (5) separate Condition II through IV licensing basis occurrences. Also that Reference 5, Page 7.2-14 Rev.42, item 1 d) specifies a maximum of 6.0 seconds (including a transport time of 2 secs) and which is confirmed by Reference 7, Table 15.1.3-1 [alongside Overpower ΔT].

The proposed T.S is non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

Item 9: Pressurizer Pressure - Low

Under these circumstances therefore, Reactor Trip on Turbine Trip is necessary to automatically terminate the event. The Licensee should review the response time used in the above calculation and provide an evaluation of its decision is respect of placing it in the T.S. under the requirements of 10CFR50.36

Item 17, [Reactor Trip on] Safety Injection Input from ESF

This description is a misnomer and should be replaced by the description proposed under Table 2.21, Item 17 of this document.

The proposed T.S. states that the response time requirement is NA (Not Applicable). This is incorrect as a separate Reactor Trip is an essential part of all ESFAs functions during which safety injection is initiated. The required information is in fact supplied in T.S. Page 3/4 3-30 Table 3.3-5, under the already revised headings proposed above, reference items 1i, 2b, 3b, 4b.

This table, under response time, should replace the description as recommended above and alongside each, reference the entry in T.S. Table 3.3-5.

The response given in the Technical Specifications (except for Manual actuation of SI) are quoted as $\leq$ 2 secs. No docketed information is available on what values were used in accident analysis, and particularly for MSLB, SBLOCA and LOCA events. The licensee should provide this information and confirm its conservatism against the T.S. value, eg. reference 5, Table 7.2.1-4 (5 of 5) and related note e. on page entitled "Notes for Table 7.2.1-4" confirms that Pressurized Low Pressure - Low Level is the first out trip of Safety Injection for the event of "Accidental Depressurization of the Main Steam System." The licensee shall explain this terminology - whether we have Reactor Trip on Pressurizer Pressure - Low which is available at the maximum power output at which this particular event is evaluated, or Pressurizer Pressure - Low (Safety Injection) and provide the associated response time to validate proposed T.S. values.

Item 21, Proposed (Reactor Coolant Pump Breaker Position Trip)

As discussed earlier, under table 2.21, Item 14, this trip is provided as an adjunct to Undervoltage - Reactor Coolant Pump Trip. The Licensee shall evaluate and propose.

Negative Steam Line Pressure Rate - High does not contain the event for the Faulted SG] then Safety injection will be activated by Containment Pressure-High.

Note: Automatic logic for realignment to SI is already provided in the T.S. in MODES 3 and 4. This MODE 4 Operability requirement for Containment Pressure-High would also facilitate re-alignment of equipment from RHR to ECCS alignment in the event of a large break LOCA under these circumstances as described in reference 8, page Q212-47a, item II.C.

The Licensee shall evaluate why his proposed T.S. is an acceptable change from the existing Licensing Basis, or include the operability requirement in his T.S. The proposed T.S. position is non-conservative.

Item 1d: Pressurizer Pressure-Low

This is the same title as used for Reactor Trip on Pressurizer Pressure-Low. This particular/ESFAS actuation is set at a lower pressure and should be described as: Pressurizer Pressure-Low [Safety Injection].

Item 1e:

The proposed T.S. for SI on Steam Line Pressure - Low is qualified in MODE 3 by a 3## which is identified on T.S. Page 3/4 3-23 as a situation in which the function may be blocked below P-12 (Low-Low T_{avg} Interlock) setpoint.

Reference 5, Table 7.3.1-3 (1 of 2) and (2 of 2) item P-1, shows the appropriate interlock for this purpose is P-11. Item P-12 of the same Table makes no provision for this proposed T.S. position.

However, reference 5 figure (6 of 16) does not use the same manual block (at P-11) for Pressurizer Pressure - Low (SI) as for Steam Line Pressure - Low (SI) (and implementation of Negative Steam Line Pressure Rate) on reference 5, Figure (7 of 16). The Licensee is required to confirm that no parameter other than the value of Pressurizer Pressure (at P-11) is used to condition the manual blocks relating to the steam line; if other parameters are used, the Licensee shall evaluate and propose. The Licensee shall also advise of other parameters which may be used to condition the manual block of Pressurizer Pressure - Low (SI).

If the Table 7.3.1-3 (1 of 2) and (2 of 2) is correct, then condition MODE 3## should be changed to condition MODE 3# which becomes the correct description.

Item 2c: Containment Pressure-High-High.

Operability is not required in MODE 4. This should be required to be consistent with the evaluation under Item 3.b.3. below.

Item 3.b3): Containment Phase B Isolation on Containment Pressure - High High

Operability of this isolation is not provided in MODE 4. The Licensee should advise why this is not necessary for safety when the previous item No.1.e.

Item 5: Turbine Trip and Feedwater Isolation

Reference earlier Item 1 in which this title for Item 5 should be more accurately described as "Turbine Trip, Trip of Feedwater pumps, Close Feedwater Isolation Valves, Close Feedwater Main and Bypass Modulating Valves. The Licensee shall clarify, evaluate and propose. Lack of accuracy n be non-conservative with respect to the Licensing Basis.

Item 5a: Automatic Actuation Logic and Actuation Relay [to effect Turbine Trip, Feedwater Pump Trip, Closure of Feedwater Isolation Valves and Closure of Feedwater Modulating Valves]/APPLICABLE MODES 1 & 2

The Applicable Modes of this Auto Actuation Logic need to be extended down to MODES 3 and 4 to be available to respond to the Safety Injection signals which are expected from the Licensing Basis (reference later Section 3/4.5, Emergency Core Cooling Systems, under GENERAL). The proposed T.S. is non-conservative with respect to the current Licensing Basis and the Licensee shall evaluate and propose.

Item 5b: Steam Generator Water Level - High High [to effect Turbine Trip, Feedwater Pump Trip, Closure of Feedwater Isolation Valves and Closure of Feedwater Modulating Valves]/APPLICABLE MODES 1 & 2.

The Licensee should evaluate the need to extend the operability requirements of this functional unit from current MODES 1 and 2 down to and including MODE 4. The determining factor may be the availablity of Main Feedwater Pumps during these MODES. Plant Operating Procedures which permit Main Feedwater Pumps to be available can cause An Excessive Heat Removal Due To Feedwater System Malfunction and/or Steam Generator overfill unless Safety Related isolation at the Main Feedwater [containment] isolation valves is incorporated into the T.S.

The Logic of reference 5, figure 7.2.1-1, (13 of 16), revision 34, involving signal inputs: Steam Generator Hi-Hi P-14, Safety Injection, Reactor Trip P4, and Low T_{avg} would need to be carefully reviewed, especially since there is currently little or no Safety Related Reactor Trip Protection in MODES 3 through 4 so that reactor trip P4 may not be available in conjunction with Low T_{avg} (during cooldown) to effect Feedwater Isolation, and Closure of Modulating Valves, as an inbuilt protection against such circumstances.

The proposed T.S. does represent a non-conservative position in respect to the Licensing Basis, as there is no prerequisite that Main Feedwater is isolated at the Containment Isolation Valves as an LCO, during MODES 3 and 4. The Licensee shall evaluate and propose.

Item 5c (Proposed): Safety Injection [to effect Turbine Trip, Feedwater Pump Trip, Closure of Feedwater Isolation Valves and Closure of Feedwater Modulating Valves]/Applicable MODES, PROPOSED AS 1, 2, 3 and 4.

This trip is relocated from Functional Unit 1 to Functional Unit 5 in accordance with our earlier reviews under Item 1C and Item 5.

Item 7.c.2): Start Turbine Driven Pumps:

Should be operable in 4. Although not capable of operating at lower temperatures of MODE 4, and MODE 5, it should nevertheless be available for use to counter consequences described in "General" above, including a station blackout.

Item 7.d): Auxiliary Feedwater Suction Pressure Low:

This proposed T.S description of a functional unit is invalid. The Functional Unit to be provided is:

d) Autor tic Re-alignment of Suction Supply [This is the functional unit], on

Low Auxiliary Feedwater Suction Pressure [This is the parameter causing the change]

Operability requirements should identify how many AFW pumps are required to be "tripped" deficient in suction, to effect re-alignment.

The licensee should identify those instrument/control channels, and particular engineering alignments, which result in a re-alignment of redundant AFW supplies to the only safety-related supply available, from the Nuclear Service Water Pond, and define related operability and surveillance requirements. The mixed nonsafety and safety-related supplies on the McGuire units make it necessary to separately define and T.S. those safety-related elements, under 10 CFR 50.46: see reference 14, page 10-2.

Applicable Modes in the current T.S. is limited to 1, 2 and 3. The licensee shall evaluate why this should not be extended to MODES 4 and 5 to meet the FSAR requirements described in "General" above.

Item 7.e: Start Motor-Driven Pumps (by Safety Injection)

Applicable Modes have not been identified. NRC proposes MODES 1, 2, 3 and 4 and 5 to meet the requirements of Item 7: General, discussed earlier.

Item 7.e: Start Turbine-Driven Pumps (by SI)

This functional unit proposes that the Turbine Driven AFW pumps are started by the SI signal. This conflicts with reference 5, Fig. 7.2.1-1 (15 of 16) I&C system Logic Diagram where the initiation of the turbine driven pumps on SI is not shown. Also, in a like manner, with related section 7.4.1.1.1.1. and reference 22, section 10.4.7.2.2.6. Also see reference 14 Section II.E.1.2 page 22-41. It is now noted that the recent T.S. has been corrected to show that the Turbine Driven AFW pump does not start on Safety Injection.] The Licensee shall clarify.

Item 10.a)a.: Pressurizer Pressure P-11:

Applicable MODES are 1, 2, 3.

Explain the consequences of this non-operability in MODE 4 on availability of dependent protective actions, e.g., main steam line isolation, which is considered under Item 4.b above. If main steam isolation is negated, it should be restored to conform to Regulatory Protection Requirement. The Licensee shall evaluate and propose.

Concerning P-11 Interlock and AFW Pumps.

The basis provided on proposed T.S. Page B 3/4 3-2 states that:

"P-11 (i.e., on system pressure increasing to P-11 valve) ---- Defeats the manual block of the motor driven AFW pumps on trip of the main feedwater pumps and Low-Low Steam Generator level."

The following information provides the current Licensing Basis on the particular proposed interlock P-11 in respect of AFW Pumps:

The Table 3.3-3, Item 7.c.1, in reference 5, for start of motor driven AFW pumps, does not provide for the above condition.

The P-11 interlock and its provision for automatic defeat [above P-11 setpoint] do not appear in reference 5, Table 7.3.1-3. Rev-35, Interlocks for ESAS and Figure 7.2.1-1 (15 of 16), revision 34, I&C Logic Diagram.

Reference 5, Section 7.4.1.1.6 describes this action under "Bypasses and Interlocks" and that whenever it is present, an alarm exists in the Control Room. This allows the operator to stop AFW pumps during shutdowns.

Supplement No. 5, reference 15, page 22-22 evaluates the use of the P-11 interlock as described in the above Basis and concludes that the situation is acceptable. However, the basis for the SER Supp 5 conclusion was that a possible steam line rupture or feedwater line break were not likely to occur in the proposed MODES when the P-11 is in effect. This is a mistake, all the earlier work of this review has disclosed that the premise of these events being not likely to occur has been rejected for these MODES 3 to 5, and detailed attention has been given to their possible occurrence together with the possibility of Auto Initiation and the consequences of automatic protective action. Where the P-11 lockout has been present on other protective actions, the consequences have been fully evaluated. There has never been a related evaluation on the absence of auto-initiation of motor-driven AFWS as now proposed.

If the Licensee wishes to pursue this he should evaluate all the events considered in the FSAR below the P-11 setpoint with manual initiation of MD AFW and making due allowance for all the relative reduced and changed protections available and the time frames which must allow for all other actions, e.g., isolation of a ruptured SG is expected to take 30 mins, see reference 7, section 15.4.2.2.2 page 15 4-13a, Revision 38. Further, the detailed review of this T.S. has been based on this availability.

Containment Isolation Valves is included in the T.S., this proposed T.S. must be considered non-conservative with respect to Regulatory Requirements.

Item 11 proposed:

There is a need to add a new Functional Unit not addressed in the current T.S., but which is a part of ESFAS.

This is:

"Close All Feedwater Isolation Valves" and "Close the Feedwater Main and Bypass Modulating Valves"

See reference 5, Figure 7.2.1-1 (13 of 16) revision 34 for the related unique control logic.

This Function is initiated by:

11a. Reactor Trip P-4, and Low Tavg.
11b. Reactor Trip P-4, and Steam Generator Level - High High P-14.
11c. Steam Generator Level - High High P-14 (see 5 above)
11d. Safety Injection (See 5 above).

Operability for 11a would be in accordance with 10c (above) and later evaluation under Table 3.3-4 Item 11a (Proposed). Operability for 11b would be in accordance with the evaluations in 10c and d above.

Operability for 11c and 11d would be by reference to items 5, 5abc.

TABLE 3.3-3: TABLE NOTATION

The uncertainty of the notation under ## is discussed in Item 1e earlier. Please amend as required in accordance with the related resolution.

Please discuss the logic of the values in reference 18. A Trip Set Point of a negative rate of 110 psi with an allowable value of 100 psi (both with a time constant of 50 psi) would provide that an earlier isolation of the MSIVs is less conservative, and this is not so for the MSLB event. The expectations are that negative rate for the allowable value would be higher than for the Set Point. Please clarify.

Further, the same reference 18 Table 3-4, column 12, states under notation (5) that this value is not used in the safety analyses. Since this ESFAS signal provides Main Steam Valve Isolation on Main Steam Line Break below the P-11 block point (instead of by Steam Line Pressure - Low) please describe how the plant is otherwise protected through the proposed T.S. Otherwise, please provide analyses which show that the plant is protected by this proposed setting under proposed T.S. requirements. This item is related to our other concerns on Technical Specifications on Boration Control under earlier Section 3/4.1.1 Boration Control. The proposition that this value is not used in Safety Aanlysis is non-conservative. The Licensee shall evaluate and propose.

Item 5: The description of this Functional Unit should be revised and clarified to our recommendations under Table 3.3-3, Item 5.

Item 5c: Proposed new item as "Safety Injection"

This should be included in accordance with the evaluation under Table 3.3-3, Item 5c)

Item 6a & b. Containment Pressure Control System

The licensee should provide the basis for these Set Points and Allowable Values.

Item 7(c): Steam Generator Water Level - Low-Low

The licensee should respond to our concern under Table 2.2-1, item 13.

Item 7(d): Auxiliary Feedwater Suction Pressure Low

The description should be revised as proposed under our earlier Table 3.3-3 item 7d. Provide the basis for the values given.

Items 7c(1) and (2): Concerning start of Motor Driven and Turbine Driven Pumps

This technical specification provides that the motor-driven AFW Pumps start on low-low in one SG whereas the turbine driven pumps require low-low in two SGs. This appears to be in conflict with the accident evaluation in the Licensing Basis FSAR as elaborated below. [This however is not conflict with the Instrumentation & Control Logic of the FSAR.]

Item 8: Automatic Switchover to Recirculation

The Licensee shall provide the basis for the set point values of the RWST levels specified. What are the allowable values for [drift and] total channel errors and the related Safety Analysis Limit.

Item 9: Loss of Power

Confirm the bases for the set points and allowable values specified.

Item: General

The Licensing Basis FSAR, reference 7, Section 15.2.9 under LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES describes a set of Reactor Protection System and Engineered Safeguards Features Actuation Responses for the Plant, to ensure its safety. Why is this particular set of ESFA's Functional Units and related Instrumentation Set Points not provided in this item under Table 3.3-4?

Absence of this information makes the proposed T.S. non-conservative. The Licensee shall evaluate and propose.

Item 10a: ESFAS Interlock Pressurizer Pressure, P-11.

Actuation of this interlock substantively reduces ECCS protection against Conditions II, III, and IV Accidental Occurrences.

The FSAR has analyzed the consequences of this reduced level of protection for a limited number of these occurrences and this has been based on a system pressure of 1900 psig; Reference 8, page Q212-47, item 212-75 1A. Why then is a trip set point of $\leq$1955 psig used. This set point value should be below 1900 psig with appropriate allowances for drift and channel errors to the limiting value used in the Safety Analysis of 1900 psig. The current specification is non-conservative with respect to the Licensing Basis FSAR & therefore not in accordance with 10 CFR 50.36. The licensee shall provide a safety evaluation for the difference, for approval, or restore the set point to be a valid T.S. value.

Item 10b: ESFAS Interlock T_{avg}-P_{12}.

The basis for this interlock on T.S. Page B 3/4 3-2 states that:

> "On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the steam dump system." This is not substantively consistent with Reference 5, Figure 7.2.1-1 which shows that it is the arming signal for the condenser dump valves and atmospheric dump valves which is removed and then with the exception of 3 cooldown dump valves (to the condenser). The steam generator Power Operated [atmospheric] Relief Valves (SG PORVs), are not affected: Please correct the Basis.

Reference Item 11b above, involving Reactor Trip P-4 & Steam Generator High High Level P-14.

The NRC has observed potential situations of concern involving this interlock.

NRC Safety Concern A: A review of the logic of this interlock, Reference 7, Figure 7.2.1-1, (13 of 16), Revision 42 shows that if a SG-Hi Hi occurs, Turbine Trip, Trip of MFW Pumps, closure of MFW isolation and control valves occur, but the reactor is not tripped if the Nuclear Power Level is below P-8 (48% Power Level), Reference 7, Figure 7.2.1-1, Revision 42, (18 of 18). This would then cause another occurrence which would be effectively a loss of main feedwater to the reactor at a nominal power level of 48%.

NRC Safety Concern B: The existing FSAR, Reference 7, Section 15.2.10.1, Revision 15, shows that a feedwater malfunction at full power is not terminated by a neutron Flux Power trip, but by a SG-Hi Hi (i.e., P-14) signal initiating Turbine Trip, Trip of MFW Pumps, Closure of MFW Isolation and MFW modulating valves. Turbine Trip will trip the reactor (if initial power level is above P-8). However, if the feedwater malfunction is initiated at zero power FSAR, Reference 7, Section 15.2.10.2, "Results," first paragraph, the consequences are a rapid increase in nuclear power which will cause a reactor trip from the neutron flux low power, 25%, setpoint, and 35% (Limiting Safety Value in Analysis) and hence generate a P-4 signal, but will not correct the initiating cause of the faulted main feedwater control system until SG-Hi Hi level is subsequently initiated and effects closure of MFW isolation valves. Whereas the FSAR evaluates the first event of this sequence by reference to the event of "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition," the FSAR provides no evaluation of the subsequent event including the DNBRs resulting from any restoration of reactivity before SG-Hi Hi ultimately effectively closes MFW isolation valves. This latter event from zero power can also occur at any intermediate power level, with and without automatic rod control, and there is currently no analysis which evaluates the worst case.

NRC Safety Concern C: The licensee has provided no information on "Safety Analysis Limits" that would be applicable to Permissive P-8 in evaluating the above events. If the allowance is ultimately of the same order as for the Power Range, Neutron Flux - High and Low Set Point Trips, i.e., approx. +10 percentage point, then Safety Concerns A and B could be occurring at up to 58% power level.

In respect of NRC Safety Concerns A, B, and C above, we consider the proposed T.S. in respect of the related permissives and interlocks to be non-conservative with respect to Regulatory Requirements. The licensee should review the safety consequences of each of these potential NRC concerns and respond with a safety evaluation with proposed changes to the T.S. as appropriate. This could be considered a Generic Issue.

General: In view of the consequences of the bypass of reactor trip on turbine trip below P-8 for the events protected by trip of turbine on

Generator Hi Hi Level to trip the MFW pumps, and together with existing Reactor Trip to provide Main Feedwater Isolation. Or, is it necessary to depend on an earlier "Isolation of Main Feedwater" from the combination of the existing reactor trip P-4 signal already provided and a related Low T_{avg}.

Inclusion of the P-4 and Low T_{avg} interlock into the T.S. would provide more reliability in protection for this event in conformance with the diversity criteria of 10 CFR 50 Appendix A, GDC Criterion 22 in support GDC 20. Without this, there is no diversity for protection from this continuing event. The proposed T.S. should require T_{avg} Low to be incorporated into the T.S. to meet the above Regulatory Criteria. The licensee shall evaluate and propose.

The licensee shall evaluate this issue with our concerns expressed under Table 3.3-4, Item 11 proposed, Reference Item 11(b) above, NRC Safety Concerns B and C to which this is directly related.

The presence of Low T_{avg}, without T.S. considerations of Set Point, Maximum Errors, Channel Reliability, Applicability MODES and Action Statements raises concerns about the consequences of a single failure. For example, a failure low, remaining undetected, could combine with a Reactor Trip from full power to close Main Feedwater [containment] Isolation valves and Main Feedwater Modulating valves and cause a more severe transient than would otherwise be necessary. The Licensee should evaluate the consequences of single failure on appropriate Conditions II, III, and IV Occurences, and propose as necessary.

Ite Reference 7, Section 15.2.14, page 15.2-38, Revision 43, which is the Accident Analysis for "Inadvertent Operation of ECCS During Power Operation," states that:

Spu ous ECCS operation at power could be caused by operator error or a false electrical actuating signal. Spurious actuation may be assumed to be caused by any of the following:

1. High Containment pressure

2. Low pressurizer pressure

3. High steam line differential pressure

4. High steam line flow with either low average coolant temperature or low steam line pressure.

Please explain the signals 3 and 4 since they do not appear in the TABLE 3.3-4 just reviewed, nor do they seem to appear in the Logic Diagrams of the Licensing Basis in the FSAR to reference 5. The Licensee shall evaluate and propose.

Hodges

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

Item 2a: Initiation of Safety Injection by: Containment Pressure-High.

A value of ≤ 27 secs (without offsite power) is given.

Reference 5, page 7.3-8 shows that initiation time of ESFAS from this source is a maximum of 1 sec.

No events in Reference 7, Section 15, have been directly analyzed using this sensor as the prime initiator above the P-11 interlock although it is relied upon for diverse protection. However, it is the only automatic initiation of Safety Injection protection below [P-11]. Other events dependent upon a SI generating signal, particularly circumstances descibed under items 3a and 4a below, shows safety analyses limits of ≤ 12 secs. (with offsite power) and ≤ 22 secs (without off site power).

At this time, the proposed T.S. value is less conservative than others used in Safety Analysis. The licensee shall evaluate this difference and propose accordingly.

Item 2b: Initiation of "Reactor Trip (From SI)" by Containment Pressure-High

The descriptor (From SI), should be deleted as it is incorrect.

The response time is give is ≤ 2 secs and this different from the FSAR, Reference 5, page 7.3-8 which gives a maximum time of 1 sec.

This value is less conservative than the FSAR and the licensee shall evaluate and propose accordingly.

Item 2c: "Feedwater Isolation" from Containment Pressure-High

The response time is given as ≤ 9 secs.

Reference 5, page 7.3-8 shows that initiation of ESFAS from this source is a maximum of 1 sec.

Table 3.6.2 of the T.S. provides isolation times of ≤ 5 secs for main feedwater containment isolation and ≤ 10 secs for main feedwater to Auxiliary Feedwater Isolation. A total time to isolation of MFW, from Containment Pressure-High, of ≤ 11 secs seems appropriate to available equipment.

There would then be a conflict between the response time of ≤ 9 secs in the proposed T.S. and the potential value of up to 11 sec from other licensing basis information.

No event in Reference 7, Section 15.1 through 4. uses this particular isolation in time Analyses. However, this is a important factor for containment integrity during a Main Steam Line Break in containment. The value used as the Safety Analysis Limit shall be provided by the licensee, Reference 5, page 7.3-8 shows that initiation of ESFAS from this source is 1 sec.

No other information is available on Safety Analysis Limits because, contrary to Regulatory Requirements, this value has not been used in the Safety Analysis of the FSAR in respect of AFW supplies. In other sections of this review, the licensee has been asked to re-evaluate Safety Analyses to recognize this fact. Parallel with this, the licensee shall identify the Actual Safety Analysis Limit to be used for this response, compare with the proposed T.S., and repropose as appropriate. Any Occurrences required to utilize Nuclear Service Water must be considered non-conservative with respect to these values currently presented in the FSAR to Reference 7, Section 15.

Item 2h: Initiation of Component Cooling Water from Containment Pressure-High

This response time is given as $65^{(3)(3)}/76^{(4)(2)}$ secs.

The description of superscript 2 under Table Notation on T.S. Page 3/4 3-33 is incomplete. The licensee shall propose an accurate description of these circumstances including its dependence on Nuclear Service Water; the licensee should confirm that this cooling water supply information is for this safety related service.

Reference 5, page 73-8 shows the initiation of ESFAS from this source is 1 sec.

No other information is available on Safety Analysis Limits used in the FSAR. The licensee shall provide this information for related Conditions II, III, and IV Occurrences for both on-site and offsite power. This information shall e evaluated and the licensee shall propose. At this time, considering the non-conservative circumstance with NSW AFW supply, it must be presumed that any Occurrence required to utilize the Nuclear Service Water must be considered non-conservative with respect to the values currently presented in the FSAR, Reference 7, Section 15.

Item 2i: "Start Diesel Generators" from Containment Pressure-High

A response time of $\leq$ 11 secs is given.

Referenc~ 5, page 7.3-8 shows that initiation of ESFAS from the source is a maximum of 1 sec.

No evaluation in Reference 7, uses this sensor as the prime initiator above the P-11 Interlock, although it is relied upon for protection above, and directly for protection below [P-11]. Other events dependent upon a SI generating signal particularly, items 3a & 4a below, show safety analysis limits of $\leq$ 10 secs for this value.

In respect of current safety analyses limits, therefore, it appears that the proposed value is less conservative than the Safety Analysis Limits. The licensee shall evaluate and propose.

Item 3c: "Feedwater Isolation" From Pressurizer Pressure-Low (SI)

The proposed T.S. is $\leq$ 9 secs.

Reference our comments and requirements under 2.c. above.

Item 3d: "Containment Isolation - Phase A" from Pressurizer Pressure-Low (SI)

The proposed T.S. is $\leq 18^{(3)}/28^{(4)}$ secs.

Reference our comments and requirements under 2.d. above.

Item 3e: "Containment Purge & Exhaust Isolation" From Pressurizer Pressure-Low (SI)

The proposed T.S. is NA.

Reference our comments and requirements under 2.e. above.

Item 3f: "Auxiliary Feedwater" Initiation by Pressurizer Pressure-Low (SI)

The licensee proposes NA (not applicable).

Safety injection logic closes the main feedwater isolation valves for every event in which SI is initiated (reference earlier sections of this review Table 3.3-4, proposed item c). Therefore, every such event initiated by a SI initiator must be analyzed with a restoration of AFW and a related response time.

It is outside the licensing basis, not to a propose a value for this response time. This T.S. value is therefore non-conservative; the licensee shall evaluate and propose.

Item 3g: "Nuclear Service Water System" Initiation from Pressurizer Pressure-Low SI

The T.S. value is given as $76^{(1)}/65^{(3)}$ secs.

Our comments on $65^{(3)}$ are as for our earlier 2g.

With respect to superscript$^{(1)}$ on 76; why is this different to Containment Pressure High which is $76^{(3)}$ when the concomitant SI signal generates the same equipment requirements. Superscript $^{(1)}$ now provides for SI and RHR pumps whereas $^{(3)}$ did not. Also, superscript $^{(1)}$, if it is to be used should include Isolation and Start of Nuclear Service Water System (NSW).

Reference our comments and requirements under earlier 2g.

Item 3: General

The licensee is to evaluate each of his superscripts $^{(1)}$, $^{(2)}$, $^{(3)}$ and $^{(4)}$ and ensure that they are complete, accurate and consistent with all the related ESFAS initiating signals and functions.

Item 4d: "Containment Isolation - Phase A" on Steam Line Pressure-Low

The proposed T.S. is $\leq 18^{(3)}/28^{(4)}$ secs.

Reference our comments and requirements under 2d. above, modified in that proposed T.S. times appear feasible with the additional delay of 1 sec.

Item 4e: "Containment Purge and Exhaust Isolation" on Steam Line Pressure-Low

The proposed T.S. is NA.

Reference our comments and requirements under item 2d. above.

Item 4f: "Auxiliary Feedwater Pumps" initiated by Steam Line Pressure-Low

The proposed T.S. is NA.

Reference our comments and requirements under 3f. above.

Item 4g: "Nuclear Service Water" initiated on Steam Line Pressure-Low

The proposed T.S. is $\leq 65^{(3)}/76^{(4)}$ secs.

Reference our comments, requirements, and remarks under 2g., 3g., and 3 General above.

Item 4h: Steam Line Isolation on Steam Line Pressure-Low.

The proposed TS value is $\leq$ 9 secs.

Reference 5, page 7.3-8 states that the maximum allowable times for generating steam break protection are (1) from steam line pressure rate, 2 secs, and (2) from steam line pressure-low, 2 secs. Further, Reference 7, page 15.4-6 states that the fast acting steam line stop valves are "designed so close in 5 secs...". A minimum closure of 7 secs seems likely.

For actual safety analysis limits, Reference 7, Table 15.4-1 (1 of 4) and 15.4-1 (2 of 4) both show a difference of seven (7) secs between arriving at the "Low Steam Line Pressure Setpoint" and "All main Steamline Isolation Valves Closed." [In the case of Feedwater System Pipe Rupture]

The proposed TS value of $\leq$ 9 secs is therefore greater than the Safety Analysis Limit.

The proposed TS must therefore be considered less conservative for this event. The licensee shall evaluate and propose.

Item 4i: "Component Cooling Water" Initiation by Steam Line Pressure-Low

Proposed T.S. value is $65^{(2)(3)}/76^{(2)(4)}$.

Reference our earlier comments and requirements under 2h and 3h. above.

The licensee shall identify the Safety Analysis Limits used for this Steam Line Isolation, including the MSLB in containment, evaluate against the proposed T.S. value and propose as appropriate. Until such time, the current value appears non-conservative.

Item 6a: Turbine Trip on Steam Generator Water Level-High High

The proposed T.S. is NA, i.e., not applicable.

Reference the licensee to our comments under Table 3.3-2, Item 16 where it is shown that it is used within the Licensing Basis.

The proposed position is non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose in accordance with our review under Table 3.3-2, Item 16.

Item 6b: "Feedwater Isolation" Initiated by Steam Generator Water Level-High High

The proposed T.S. is $\leq$ 13 secs.

Reference 7, Table 15.1.3-1 shows that "High Steam Generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip" is based on an ESFAS time delay of 2.0 seconds.

Table 3.6.2 of the T.S. provides isolation times of $\leq$ 5 secs for main feedwater containment isolation and $\leq$ 10 secs for main feedwater to Auxiliary Feedwater Isolation.

A total time to isolation of MFW of $\leq$ 13 secs seems appropriate to available equipment.

However the current safety analysis depending on this response time is that for the Excessive Cooldown occurrence under Reference 7, page 15.2-28, and for this, no value is quoted for isolation of main feedwater which is the initiator of the event. However, Figure 15.2.10-2 shows that with initiation of the event caused by one faulty control valve, it takes 32 secs to reach the SG-High-High Level with a mass increase of 35% of initial, and thereafter does not increase further. This implies zero closure time. Since it is expected to take another 13 secs to actually isolate, we could assume an additional mass increase of another 13% to give a total of approx. 1.48 the initial value.

The above additional Main Feedwater level can affect the consequences of the event at power, if there has been a trip, with a potential for power restoration and/or overfill of the S-G to cause water ingress into the main steam lines. Additionally, it can have consequences of potentially larger importance for the event occurring from zero subcritical power.

Reference also our concerns under item Table 3.3-4, item 11b and 11a above.

The licensee shall evaluate the related concerns, including the extended MFW valve isolation times, to determine their safety significance, and

Item 11, below. If this is confirmed at from 65 to 70 secs, or any longer time than used as the existing Safety Analysis Limit in the FSAR, then acceptable re-evaluation of all Conditions II, III, and IV occurrences involving AFW supply, are required by 10 CFR 50.36.

Our current evaluation is that the response times in the proposed T.S. are non-conservative in respect of Regulatory requirements.

Item 8: "Steam Line Isolation" on Negative Steam Line Pressure Rate-High

Proposed T.S. value is $\leq$ 9 sec.

Reference 5, page 7.3-8 states that the maximum allowable time for generating the ESFAS MSIV isolation signal from a Steam Line Pressure Rate circumstance is 2 secs, the same as for item 4h. above.

Our comments and requirements therefore are the same as under item 4h.

We appreciate that this signal is generated at below P-11, but with the existing proposed Boration Control T.S. we must continue to evaluate this value as non-conservative.

The proposed T.S. value is greater than the Safety Analysis Limit of seven (7) secs and must be considered less conservative for this event. The licensee must evaluate this difference and propose.

Item 11: "Automatic Re-alignment of AFW Supply on Low Suction Line Pressure" [The existing description should be changed to more accurately state this action]

Proposed T.S. value is 13 secs.

Note our comments under 7a. and 7b. above. Although this response time may be in accordance with current plant engineering, it is not in accordance with the existing Safety Analysis Limit for Auxiliary Feedwater Supply which, on current information, has pre supposed no such transfer time. If a tank has been lost because of seismic action, we cannot assume a residual 15 secs supply at this time.

At this time, until the evaluation of 7a. and 7b. above is completed, we must evaluate this delay as non-conservative with respect to currently used Safety Analysis Limits which in themselves are non-conservative with respect to Regulatory requirements.

The licensee will evaluate and propose.

Item 12: "Automatic Switchover to Recirculation" on Low RWST Level

Response time proposed as $\leq$ 60 secs

The licensee shall provide the bases for this value and evaluate against this $\leq$ 60 secs, and propose as necessary.

Item 15: Loss of Power

Item 15: General

Our review comments under item 13 "Station Blackout" are fully applicable to this item with the related conclusion that:

The absence of most of the information on Functional Units and related Response Times required to Protect the Facility on Loss of Power makes the proposed T.S. non-conservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

Item [Foot] Note. Response time for Motor-Driven Auxiliary Feedwater Pump Starts on All SI signals.

This is proposed as $\leq$ 60 secs

Reference our earlier comments for its inclusion in Items 2f., 3f., and 4f. above together with the necessary Licensee Actions.

Reference our earlier comments under 7a. and 7b. above together with the necessity for licensee action.

At this time, these values are non-conservative with respect to Regulatory requirements and the licensee must evaluate and propose.

Item: Table 3.3-5, TABLE NOTATION on T.S. Page 3/4 3-33

These notations 1, 2, 3, and 4 must be expanded to include Component Cooling Water System Isolation and Pumps, Nuclear Service Water System (NSWS) Isolation & Pumps, and AFW re-alignment to NSWS and alternate sources as necessary. This will also enable verifiable consistency with the Notations used in the table.

See our comment under items 2g., 2h., 3g., 3h., 4g., and 4i. above.

Notation 2 of this Table states that:

(2) Valves 1KC305B and 1KC315B for Unit 1 and Valves 2KC305B and 2KC315B for Unit 2 are exceptions to the response times listed in the table. The following response times in seconds are the required values for these valves for the initiating signal and function indicated:

2.b $< 30^{(3)}/40^{(4)}$
3.b $\leq 30^{(3)}$
4.b $\leq 30^{(3)}/40^{(4)}$

Since the functions 2b, 3b and 4b are all Reactor Trip functions, please explain.

Since these descriptors are apparently incorrect, provide the correct descriptors.

Section 3/4.4 REACTOR COOLANT SYSTEM

Section 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

Item: GENERAL

G.1 INTRODUCTION

Concerning RCS Operability requirements, in MODE 3-5:

We refer to our earlier discussions & licensee requirements - and especially under Section 3/4.1.1, T.S. Page 3/4 1-1, 2 & 2a on Boration Control, T.S., Page 3/4 1-20 & 1-21 concerning SHUTDOWN AND CONTROL ROD INSERTION LIMITS and TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION - generally, including more particularly items 2-21 (selected) and items 12, 14, 15 and 21.

Under our item T.S. TABLE 3.3-1, items 2, 5 & 6 et al, the licensee has been required to "Provide an anlaysis and evaluation of the consequences of Applicable Condition II, III and IV Occurrences, in MODES 3 through 5, for an appropriate set of Technical Specification requirements to ensure Conformance to Acceptable Regulatory Criteria, and from this establish an appropriate range of Reactor Trip System Instrumentation to Safety Related Requirements. This evaluation shall be undertaken in conjunction with our concerns for current technical specifications under section 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION of this review.

As part of this review, and as a safety justification for our concerns, we require inclusion of the following Occurrences and Considerations in the program, and as early determinants of our proposals in respect of RCS Loop Operability requirements in MODES 3, 4 and 5 (with loops filled).

G.2 DISCUSSION

Item: CONSIDERATION

A number of factors determine our concern:

G.2.1 The increased boron concentration discussed under Section 3/4.1.1 of this review.

G.2.1.1 Increases shut down margin at temperatures above 200°F, and thereby reduces the severity of any occurrences giving a return to power, but only after reactor trip. Further the T.S. proposed by the licensee does not include the increased boron concentration and RCS Operability requirements are judged against those circumstances.

G.2.1.2 Because increased shutdown margins are available, in MODES 3, 4 and 5, the licensee may now increase the level of withdrawal of all movable control assemblies and still remain within the unchanged T.S. condition of the allowable reactivity condition, keff of $\leq$ 0.99. Consequently, it does not benefit those Occurrences initiated by fast positive reactivity excursions in which maximum power levels ultimately reached are substantively determined by given Response Times operability of Reactor Trip from SI in this mode and offers no Safety Evaluation for the proposed change. Reference our review under Table 3.3-1, Item 17.

The proposed T.S. is not in conformance with the Licensing Basis, and is nonconservative. The licensee shall evaluate and propose.

G.2.4 In MODE 3, from P-11, to MODE 5, for events initiating SI, the plant is engineered and can be operated so that only one automatic trip of the reactor may be available; that from containment pressure-high.

On the above bases, plant engineering and operations would not be in conformity with regulatory requirements. The Licensee shall evaluate and propose.

It may be possible for the plant to be operated in a manner to conform by not manually blocking the Main Steam Line Pressure-Low Trip [at P-11] but constraining this blockage to a point at which SG pressure during cooldown is within an acceptable error band of the related Set Point Value. Under these circumstances, two (2) diverse automatic protections on reactor trip may be available.

In addition the proposed T.S.s do not require operability of the Reactor Trip/ESF channel in this phase of operations below MODE 3 [at P-11], to MODE 4 even though this is engineered into the Facility. No Safety Evaluation of this omission is provided. The FSAR assumes Safety Injection Protection in MODES 3 and 4. The proposed T.S. is not in accord with the Licensing Basis and is nonconservative. The Licensee shall evaluate and propose.

G.2.5 Diversity of Safety Injection to the maximum extent for related Accident Circumstances can only be retained within existing plant engineering by requiring that manual block of the Steam Line Pressure-Low be delayed until SG pressures are within an appropriate error band of the Steam Line Pressure-Low Set Point. This could be down to a temperature of approximately 485-490°F in the RCS which would be in MODE 3 before 1000 psig/425°F. (485-490°F is the saturation temperature equivalent to 565 psig + 30 psig [channel error] i.e., approximately 595 psig in the SG.

The licensee shall evaluate and propose.

G.2.6 <u>EVENTS OF CONCERN (A LIMITED SELECTION)</u>

G.2.6.1 OCCURRENCES WITH RAPID REACTIVITY INCREASE

Concerning "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Sub-Critical Condition."

Current Docketed Analysis in reference 7, section 15.2.1, page 15.2-2 is based on four operating loops. This event is possible down to and including Mode 5. Current FSAR analysis trips the reactor on Power Range, Neutron Flux-Low Set actuation is manually blocked on low steamline pressure and low pressurizer pressure."

This position gives no measure of the resulting shutdown margins and/or power level and, the consequences of a stuck rod, with only 2 RC loops operating instead of four. It is conceivable that two loop operation may be less conservative than either 4 RCPs continuing to operate or 4 RCPs tripped on Safety Injection, due to an increased cooldown in the core due to circulation (compared to the tripped case) but a much decreased core flow rate to handle the event. The potential short term consequences of bulk voiding and loss of circulation in the non-operable loops cannot be ignored.

If during cooldown, an MSLB cools the RCS down to 212°F e.g., the residual shutdown will be at 1% delta k/k whereas the proposed T.S. margin at Zero Power according to T.S. Page 3/4 1-1 was 1.6 delta k/k. Please cla ify, and at what condition during cooldown the 1.6% delta k/k is reached.

Given the circumstances that the "Operating Instructions" described above are not a part of the proposed T.S., any T.S. allowing operability of less than 4 RCS Loops in MODE 3 would be in non-conformance with the current Licensing Basis Safety Analysis in the FSAR in a non-conservative manner, and the licensee would be required to evaluate and propose.

For this licensing basis event, from Zero Power, Reactor Trip does not occur on Power Flux Trip, but on Pressurizer Pressure-Low (SI) (above P-11) [reference our required confirmation of this in an earlier item] so the Power Flux Trip is not required to be Operable.

At less than P-11, these circumstances are changed for the MSLB, and Reactor Trip does not occur until Containment-Hi is achieved, for a break inside containment.

For a break outside containment, however, high negative steam rate isolates main steam isolation valves only, but their is no Safety Injection, no Reactor Trip (on SI), and under the exisiting proposed T.S. no safety related Reactor Trip System Instrumentation of any nature to Trip the Reactor and Insert the movable control rods to benefit from potentially increased available shutdown margin. In addition to all this, the licensee proposes that MSIV closure times under these conditions in Not Applicable.

Given the circumstances of the proposed T.S., and T.S. allowing OPERABILITY of less than 4 RCS Loop in MODE 3 under these circumstances would be in nonconformance with the current Licensing Basis FSAR in a nonconservative manner, and the licensee would be required to evaluate and propose.

Additional events which exhibit a rapid cooldown and depressurization of the RCS; are:

a) Accidental Depressurization of the main steam system at no load, (reference 7, page 15.2-35, revision 36).

b) Minor Secondary System Pipe Breaks [at no load]; reference 7, page 15.3-4, revision 27).

Additional events of a similar nature to the SBLOCA events include:

a) Accidental Depressurization of the Reactor Coolant System (reference 7, page 15.2-33, revision 7).

b) Steam Generator Tube Rupture (reference, page 15.4 - 13a, revision 38).

c) Rupture of a Control Rod Drive Mechanism Housing at Zero Power (reference 7, page 15.4.6, revision 42).

Both events, a) and b), are analyzed in the Licensing Bases at Full Power, and use Pressurizer Pressure-Low as a first reactor trip. At zero power, with current proposed T.S. this reactor trip is proposed as Not Operable.

For event c), from Zero Power, Power Range Neutron Flux, High Set Point Trips the Reactor; Pressurizer Pressure-Low (SI) initiates Safety Injection; reference 7, page 15.4-29, revision 43, paras. 1 and 5. Whereas both these protections are proposed by the T.S. in MODE 2, they are not proposed for MODE 3 which differs from the circumstances of MODE 2 by only a marginal reduction in RCS Temperature.

The FSAR, reference 7, Table 15.4.6-1, revision 42, shows this occurrence as being the only event at Zero Power, analyzed to a smaller N° of RCPs than 4; it has been analyzed for 2 only. This is an accident with substantial but "acceptable to Condition IV occurrences" consequences in terms of fuel cladding damage and RCS overpressurization, but it required at least two RCPs to achieve that (in the Licensing Basis). Even the two RCPs required in this event are not proposed as being required for MODE 3.

The proposed circumstances in MODE 3 are clearly non-conservative with respect to the Licensing Bases. The licensee shall evaluate and propose.

Concerning the Large Break "Loss of Coolant Accident."

This is discussed in Accident Analyses in Reference 7, section 15.4.1 for a LOCA from rated power; in Reference 8, item 212.75 page Q 212.47, for a LOCA between RCS conditions of 1900 psig and 1000 psig/425°F in Hot Standbye; in item 212.90(6.3), page 212-61, for a LOCA at and less than 1000 psig/425° in Hot Standbye, and on page Q 212-61b, item 29 for a LOCA in the RHR Mode at 425 psig/350°F.

As for the Small Break LOCA, these analyses are presumably based on 4 RCS loop operation, with in general, loss of power to RCS Pumps on Safety Injection.

The large break LOCA analyses used the Topical Report WCAP-8479, reference 7, page 15.4-1. At this time, we expect no difference in the importance of RCPs to that discussed under the paragraph commencing "Concerning Small Break LOCA" which used the W Topical Report WCAP 8356 (reference 19) and which applied to both Large and Small Break LOCAs.

operation of 4 RCS Loops, whilst on RHR, may be undesirable because of the substantial additional burden on the RHR system; so, nonoperability of all RCPs must be compensated by other controllable factors such as inserting all movable control assemblies and removing power from the Reactor Trip System Breakers, closure of Main Feedwater [Containment] Isolation valves to both Main and Auxiliary Feedwater Systems, Closure of Main Steam Isolation Valves, and Boration Control measures additional to those included in the proposed T.S. An additional available alternate action is to use, within MODE 4, a minimum set of RCS pumps (and loops) as established by Safety Analysis, to cool the plant down to effectively zero pressure (gauge) in the Steam Generators [or less if the condenser was still available] before transferring the heat sink to the RHR system. This would ensure control of Steam Line Break, and LOCA events, small and large, down to RCS conditions where RCS flows are not necessary.

The current T.S. are nonconservative in respect to the Licensing Basis in respect to these concerns. The Licensee shall evaluate and propose.

T.S. SECTION 3/4.4.1: RCS LOOPS AND COOLANT CIRCULATION

START UP (MODE 2) AND POWER OPERATION (MODE 1).

The LCO requires all [4] reactor coolant loops to be in operation in MODES 1 & 2.

The ACTION Statement requires that in the event of loss of 1 [of 4] RCS Loop in MODES 1 & 2, the licensee is required to be in at least HOT STANDBY within 1 hr.

The current Safety Analysis Limits in the FSAR, reference 7, page 15.2-16, revision 7, requires an immediate trip of the reactor to RTI & ESFAS response times in the event of loss of 1 RCS pump. Also, placement of the RCS in Hot Standby with less than one loop operable [without other compensating conditions] would be non-conservative in respect of the existing FSAR.

The Action Statement is non-conservative with respect to the current licensing basis and the licensee shall evaluate and propose.

T.S. surveillance requires verification of Reactor Coolant Loop (RCL) circulation once every 12 hours. This is unacceptable considering the Safety Analysis limits required above for loss at one pump. In the event of failure of the Low Reactor Coolant Flow Reactor Trip; the operator should respond immediately to the related Alarm to trip the reactor, if it remains. Reference to earlier work of this review will show that there is no alternate, or diverse, sensor for low flow in one Reactor Coolant Loop. Further the FSAR analysis does not provide an evaluation of the consequences of a 10 min delay by the operator on hearing the Alarm - if it has remained operable from available [3 channel] LOGIC. Additionally, the FSAR proposes no alternate trips for the reactor, with related evaluation, such as over temperature leading to Pressurizer Level-High and Pressurizer Pressure-High. The Action Statement would place the plant outside the current licensing basis for normal operation and is non-conservative with respect to that. The licensee shall evaluate and propose.

Potential

operation of 4 RCS Loops, whilst on RHR, may be undesirable because of the substantial additional burden on the RHR system; so, nonoperability of all RCPs must be compensated by other controllable factors such as inserting all movable control assemblies and removing power from the Reactor Trip System Breakers, closure of Main Feedwater [Containment] Isolation valves to both Main and Auxiliary Feedwater Systems, Closure of Main Steam Isolation Valves, and Boration Control measures additional to those included in the proposed T.S. An additional available alternate action is to use, within MODE 4, a minimum set of RCS pumps (and loops) as established by Safety Analysis, to cool the plant down to effectively zero pressure (gauge) in the Steam Generators [or less if the condenser was still available] before transferring the heat sink to the RHR system. This would ensure control of Steam Line Break, and LOCA events, small and large, down to RCS conditions where RCS flows are not necessary.

The current T.S. are nonconservative in respect to the Licensing Basis in respect to these concerns. The Licensee shall evaluate and propose.

T.S. SECTION 3/4.4.1: RCS LOOPS AND COOLANT CIRCULATION

START UP (MODE 2) AND POWER OPERATION (MODE 1).

The LCO requires all [4] reactor coolant loops to be in operation in MODES 1 & 2.

The ACTION Statement requires that in the event of loss of 1 [of 4] RCS Loop in MODES 1 & 2, the licensee is required to be in at least HOT STANDBY within 1 hr.

The current Safety Analysis Limits in the FSAR, reference 7, page 15.2-16, revision 7, requires an immediate trip of the reactor to RTI & ESFAS response times in the event of loss of 1 RCS pump. Also, placement of the RCS in Hot Standby with less than one loop operable [without other compensating conditions] would be non-conservative in respect of the existing FSAR.

The Action Statement is non-conservative with respect to the current licensing basis and the licensee shall evaluate and propose.

T.S. surveillance requires verification of Reactor Coolant Loop (RCL) circulation once every 12 hours. This is unacceptable considering the Safety Analysis limits required above for loss at one pump. In the event of failure of the Low Reactor Coolant Flow Reactor Trip; the operator should respond immediately to the related Alarm to trip the reactor, if it remains. Reference to earlier work of this review will show that there is no alternate, or diverse, sensor for low flow in one Reactor Coolant Loop. Further the FSAR analysis does not provide an evaluation of the consequences of a 10 min delay by the operator on hearing the Alarm - if it has remained operable from available [3 channel] LOGIC. Additionally, the FSAR proposes no alternate trips for the reactor, with related evaluation, such as over temperature leading to Pressurizer Level-High and Pressurizer Pressure-High. The Action Statement would place the plant outside the current licensing basis for normal operation and is non-conservative with respect to that. The licensee shall evaluate and propose.

Given the circumstances of the proposed T.S., operability of less than 4 RCS loops in MODE 3, HOT STANDBY, would be in non-conformance with the current Safety Analysis Limits in a non-conservative manner and the licensee is required to evaluate and propose.

It further follows, that the proposed surveillance requirement T. S. item 4.4.1.2.3 that at least one reactor coolant loop shall be verified in operation and circulating reactor coolant at least once 12 hours is also invalid and should be changed.

The surveillance requirement, once every 12 hours, is intended to ensure not only that the system is operating, but that is is operating at process conditions which can be evaluated to show that the equipment is capable of performing its Licensing Basis Safety Functions. The proposed T.S. requirements are absent in this information; it is therefore non-conservative and the licensee shall evaluate and propose.

Surveillance requirements for the S.G. call for a level of 12% at least once per 12 hours. This is not in accordance with the Licensing Basis; this level is the S.G. Low - Low Trip Set Point. All conditions II, III and IV occurrences require in general, for this S.G. level to be at the programmed Set Point for the Zero Power Condition with automatic actuation; we have no evaluation at alternate conditions. Therefore this exlisting proposal is outside the current Licensing Basis and non-conservative. Reference our earlier comments under Item 2.1.1, Item f. The licensee shall evaluate and propose.

*This Footnote proposes that; in HOT STANDBY (MODE 3):

"*All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature."

This is a natural circulation condition; the only Licensing Basis calculation for this is the Natural Circulation calculations of reference 7, page 15.2-27, "Loss of Offsite Power to Station Auxiliaries"; but at MODE 2 Zero Power conditions with related programmed process conditions of Zero Load Pressure and Temperature in the loops. No basis is provided for ensuring that natural circulation will be safe over the range of conditions now expected in this MODE 3. Earlier considerations show that more comprehensive protections against the possibility of Condition II, III and IV occurrences must involve, in addition to isolation of all boron dilution sources, securing Reactor Trip System Breakers in the Open Position, closure of MFW isolation valves, isolation of MSIVs, and possibly an optimum boron concentration. At present, the only Licensing Basis for controlling this particular situation is the Emergency Operating Guidelines.

Given the circumstances of the proposed T.S., the proposal to de-energize 4 RCPs for up to one hour is outside the Safety Analysis Limits of the FSAR and is non-conservative with respect to that.

The licensee shall provide the reason for this requirement including the expected condition of the Facility, and then analyze, evaluate and propose.

eliminated, the safety status of the facility is outside the Licensing Basis of the FSAR in a non-conservative manner.

Each of the OPERABLE loops, whether RCS or RHR, are to be energized from separate power divisions to protect against single failure of a bus or distribution system. When the RCS systems are used, the related Auxiliary Feedwater systems are also required to be operable.

The additional requirement proposed, for two RCS loops to be operable whenever RHR loop/s are in operation, is based upon reference 8, page Q 212-55 and 56, to provide for the failure of a single motorized valve in the RHR/RCS suction line in both MODEs 4 and 5 and possible non-availability of offsite power sources. The FSAR provides, that on failure of the valve:

> "Approximately 3 hours are available to the operator to establish an alternate means of core cooling. This is the time it would take to heat the available RCS volume from 350°F to the saturation temperature for 400 psi (445°F), assuming the maximum 24 hours decay heat load.
>
> To restore core cooling, the operator only has to return to heat removal via the steam generators. The operator can employ either steam dump to the main condenser or to the atmosphere, with makeup to the steam generators from the auxiliary feedwater system. The time required to establish the alternate means of heat removal is only the few minutes necessary to open the steam dump valves and to start up the auxiliary feedwater system."

The APPLICABILITY MODE 4, is necessarily qualified by [less than 425 psig/350°F] by the LOCA analyses already referenced above under our review Section 3/4 4.1 Subsection G.2.6.3 "Concerning Large Break Loss of Coolant Accident." See reference 8, page Q 212-47.d where it is described that

> "After several hours into the cooldown procedure (a minimum time is approximately four hours) when the RCS pressure and temperature have decreased to 400 psig and 350°F."

And arising from a later revision 25, the FSAR advises on page Q 212-61b revision 29 concerning ECCS calculations in a later submittal under Revision 28 that

> "The response provided in Revision 28 addressed the subject of operator actions and ECCS availability. Consistent with the information provided in Revision 28, a postulated LOCA in the RHR mode at 425 psig RCS pressure has been assessed."

The additional Action statement that:

b. "With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective ACTION to return the required coolant loop to operation."

Item 4.4.1.4.4 (Proposed). It is proposed that an additional item be inserted which reads: "The related auxiliary Feedwater System shall be determined OPERABLE as per the requirements of T.S. 3.7.1.2 [and 3.7.1.2.a as applicable]." Current proposed T.S.s on T.S. page 3/4 7-4 are non-conservative in this matte. by not providing any operability requirements for AFW in this MODE. The licensee shall evaluate and propose.

An additional item is also required in which Atmospheric Dump Valves operability is established. The current T.S. are non-conservative in this matter; they make no provision for operability of this item (see later proposed T.S. page 3/4 7-8a). [General comment: Operability of each of S.G. water level, AFW and ATMOSPHERIC DUMP VALVES in this MODE is probably better defined under each of these items in their particular sections of the T.S. See later sections of this review as identified above.]

The FSAR addresses the consequence of a failure, closed, of the isolation valve in the RCS/RHR line; it addresses the analysis from 350°F in the RHR MODE when a bubble is present in the pressurizer. This will also be valid down to the RCS temperature at which the bubble will be established, i.e., below 300°F according to reference 19, page 52-21a, revision 33, first para. If the licensee does operate the plant so that the system is water solid between 200°F and 300°F in MODE 4, a loss of cooling could result in a potential overpressurization of the system and the reviewer is not aware of any evaluation of the adequacy of the existing Low Temperature Overpressure Protection System to accommodate that event. The licensee shall evaluate and propose.

<u>T.S. Page 3/4 4-5: COLD SHUTDOWN [MODE 5] WITH LOOPS FILLED</u>.

The current proposed T.S. provides:

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 12%.

The current FSAR requires two (2) OPERABLE RHR trains on two (2) redundant electrical buses so that each pump receives power from a different source, reference 20, Pages 5.5-24. In the event of Loss of Offsite Power, the pumps are automatically transferred to a separate emergency diesel power supply. Therefore; the current licensing basis is that 2 residual heat removal loops shall be operable. The above provision for either an RHR loop or two steam generators is therefore not in accordance with the Licensing Basis. The proposed T.S. in this respect is also non-conservative as it would necessarily require S.G. temperatures greater than 212°F (Atmos Press in SGs) which would place it outside the Cold Shutdown MODE into the Hot Shutdown MODE - which is outside the required Functional MODE.

The T.S. requirement for one RHR loop in operation and one to be available OPERABLE is currently not supportable by analysis evaluating the situation in which all RHR cooling is lost in a water solid condition; reference our power from a different source; reference 20, pages 5.5-24, revision 9. Without this requirement, the T.S. is less conservative than the FSAR and the licensee shall evaluate and propose.

Additionally, the current FSAR, reference 8, page Q 212-57, revision 25, describes that in the event of loss of flow caused by isolation of the RHR/RCS Isolation valve [and also by cessation of flow in the system]

> "The operator would be alerted to the loss of RHR flow by the RHR low flow alarm.
>
> Assuming worst case conditons (maximum 24 hours decay heat, air in the steam generator tubes, and the RCS drained to just below the vessel flange) and making conservative assumptions about the amount of water available to heat up and boil off, if the operator took no action, boiling would begin in about five minutes, the water level in the vessel would be down to the level of fuel in about 100 minutes, and the pressure would increase to 550 psi in about 40 minutes (the pressure rise could be limited to about 550 psi by opening the pressurizer power operated relief valves)."

In the event only 1 RHR loop is required to be in operation,the LCO should therefore require 2 operable Safety Related RHR flow alarms on each single operating RHR system so that the operator can respond within 10 mins to commence operation of the redundant system. However, this time frame is excessive since boiling will have commenced. It is necessary to maintain two operating RHR systems so that boiling may be eliminated on single failure. The licensee shall evaluate and propose.

Additionally, the above information defines an LCO of a minimum volume of water for the related event in which the RCS is drained to just below the Reactor Vessel flanges and which minimum volume shall be included in the T.S. as an LCO with appropriate surveillance and Action Statements. A further T.S. requirement is that any such min volume should be such that the level of water in or above the RCS loops be such as to provide acceptable flow, including NPSH conditions, over the range of temperatures expected, at inlet to the RHR pumps. Absent those required conditions from the Limiting Conditions of operation makes them non-conservative in respect to the Licensing Basis. The licensee shall evaluate and propose.

Concerning Action item b., this provides that

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective ACTION to return the required RHR loop to operation.

Further: In the event that RHR cooling cannot be restored in "sufficient" time, the FSAR states that, in the event of loss of flow caused by the single RCS/RHR motorized valve:

> "To restore core cooling, the operator would first attempt to fill and pressurize the reactor coolant system with the centrifugal charging pumps. If the system can be pressurized to the range of 400-500 psi, the

The safety basis for this was established in the FSAR, as indicated in earlier sections, and the need for safety related redundancy arises to ensure RCS integrity to Safety Related Criteria as discussed above. The current T.S. is non-conservative with respect to the Licensing Basis.

T.S. SECTION 3/4.4.2 SAFETY VALVES

SHUTDOWN (MODES 4 and 5)

The T.S. requires that:

"3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling MODE."

Reference our review comments and requirements under T.S. 3/4.4.2 SAFETY VALVES, OPERATING which are also applicable to this section. The current T.S. must be considered nonconservative with respect to the Licensing Basis. The Licensee shall evaluate and propose.

The Action statement is based (reference T.S. page B 3/4.4-2) on the premise that INOPERABILITY of the Safety Valve in Modes 4 and 5 needs to be offset by operability of pressure relief valves in the RHR systems. This is not the safety basis for Action. The safety basis is, that the Reactor Coolant Pressure Boundary has been effectively rendered inoperable requiring the operator to proceed to a cold shutdown condition with the zero pressure (gauge) in both RCS and SG systems, and related reactivity control actions to ensure that no return to nuclear power is possible. This needs to be done in a manner consistent with the nature of inoperability of the Safety Valve. The current T.S. is nonconservative with respect to the Licensing Basis; the licensee shall evaluate and propose.

Further, McGuire Units 1 and 2 do not use RHR overpressure protection of the RCS as the plant utilizes two available PORVs on the pressurizer, reset to 400 psig (reference review under T.S. Page 3/4 4-36) in the primary coolant system. In this respect, the proposed action statement is non-conservative and contrary to the Licensing Basis. The licensee shall evaluate and propose.

The Surveillance Requirements should contain the minimum discharge capacity required of this valve as defined in the Licensing Basis. They should also ensure the maintenance of satisfactory environmental conditions consistent with reliable valve operability. The licensee shall evaluate and propose.

measurable parameters "in operation" to ascertain the status of the valve so that acceptable measures can be taken.

The safety basis for the concern rests not only in the previous position addressed above, but also, that in the event of failure of control grade "pressure control devices" these valves will be challenged on the following occurrences within the Licensing Basis.

- Startup of the Inactive Coolant Loop; reference 7 Figure 15.2.6-1, revision 4
- Loss of Load Accident; reference 7, Figure 15.2.7-5, revision 38
- Loss of Normal Feedwater; reference 7, page 15.2-26, revision 7, para. 3
- Main Feedwater Line Break Accident, reference 7, Figure 15.4.2.7, revision 38
- One Locked Rotor Event; reference 7, Figure 15.4.4-1, revision 32

Safety Valve Operation could also occur on other overpressurization events if some of the early reactor trips fail to operate as expected.

In this matter, the T.S. is nonconservative with respect to Regulatory Requirements. The Licensee shall evaluate and propose. This could be a generic issue.

Surveillance Requirements should reference the documents containing the record of the Inservice Testing of the valves for inspection on a regular basis of 12 hours so that changing operating staff are kept aware of a potentially changing status on a singularly critical item.

<u>T.S. Section 3/4.4.3 PRESSURIZER</u>

<u>T.S. Page 3/4 4-9</u>

The APPLICABILITY MODES are proposed as 1, 2 and 3.

Item: Pressurizer Level:

The response of all the analyses of Condition II, III and IV events in references 7 and 8 depend upon an initial level of water in the Pressurizer which is programmed as a varying value dependent upon the Nuclear Power Level. Additionally, the response of all Condition I events which determine the most conservative set of parameters from which to start Condition II, III and IV events, are also so dependent upon this same programmed pressurizer level.

Since therefore this pressurizer level is used in establishing an acceptable outcome of these analyses in terms of the issuance of the operating license, they also represent limiting conditions of operation as defined in 10 CFR 30.46. On this basis therefore, the licensee should provide details of the programmed pressurizer level set points with allowable values consistent with the related channel errors and Safety Analysis Limits used in the FSAR, Section 15 in reference 7. The licensee shall evaluate and propose.

(2) valves to one (1) only valve, the Regulatory Requirements are not met and the plant must proceed to a cold shutdown condition with no potential for positive reactivity changes, within appropriate time frames.

The current T.S. is nonconservative in respect to Regulatory Requirements. The licensee shall evaluate and propose.

T.S. Section 3/4 4.5 STEAM GENERATORS

T.S. Page 3/4 4-11

a) S.G. Levels

A number of the Accident Analyses in reference 7 depend upon an initial level of water in the Steam Generator. A specific example is the Main Feedwater Line Rupture Event of Section 15.4.2.2.2 in which AFW auto-start signal on SG low-low level occurs 20 secs are main feedline rupture occurs; reference related Table 15.4-1, page 1 of 4].

Since this, and other events, depend upon a "programmed" water level in the steam generators for an acceptable outcome in terms of the issuance of the operating license, these water levels also represent limiting conditions of operation in respect of 10 CFR 30.46. Please provide details of such SG levels including related Safety Analysis Limits, and respond to the proposition that such values should be included as Set Point values and Allowable values in the proposed T.S. as Limiting Conditions of Operation for the facility with appropriate Action Statements. The proposed T.S. is nonconservative by their absence.

b) Steam Generator Pressures

Since Steam Generator Pressures and related Saturation Temperatures under normal steady state operation can be a significant determinant of system responses for Condition II through IV occurrences analyzed in the Licensing Basis including Section 15 of reference 7, and reference 8, please provide the values used as Safety Analysis Limits in related analyses and again respond to the proposition that such values should be included as Set Point and Allowable values as Limiting Conditions of Operation for the facility with appropriate Action Statements. The proposed T.S. is nonconservative with respect to the Licensing Basis, by their absence.

c) Please respond to the proposition that this section should also adequately identify the maximum allowable Steam Generator Pressure under Transient and Accident conditions with appropriate Action Statements. Maximum SG pressure is one of the Acceptance Criteria for safety. The current very limited basis for Steam Generator Pressure integrity is completely inadequate. Please clarify apparent discrepancy between reference 4, Table 5.5.2-1 in which the steam side design pressure for the Steam Generator is given as 1285 psig and the value quoted in the T.S. Basis Page B 3/4 7-1 at 1185 psig.

The proposed T.S. is nonconservative with respect to the Licensing Basis, by this absence.

We find no safety evaluation in the Licensing Basis for the alternate use of an RCS vent of greater than or equal to 4.5 square inches in the proposed T.S. The licensee shall evaluate and propose.

B) Between 1000 psig and 400 psig, a portion of the ECCS can be actuated automatically (containment high pressure signal) or manually by the operator. The equipment that can be energized are two RHR pumps and one charging pump. The operator would have to reinstitute power at the motor control centers or switchgear to the remaining safety injection pumps, charging pump, and the accumulator isolation valves.

C) Below 400 psig, the system is in the RHR cooling mode. The RHR system would have to be realigned as per plant startup procedure. The operator would place all safeguards systems valves in the required positions for plant operation and place the safety injection, centrifugal charging, and residual heat removal pumps along with SI accumulator in ready and then manually actuate SI."

In response to additional questions, the following information was provided under FSAR reference 8, page Q 212-61, revision 28, item 212.90(6.3); page Q 212-61a, revision 28, pages Q 212-61b, revision 29 and Q 212-61c, revision 29

"In spite of the low probability of occurrence and the fact that certain failure modes for pipe rupture do not exist during cooldown at an RCS pressure of 1000 psig, the following items have been incorporated into the station operating procedures:

1. At 100[0] psig, the operator will maintain pressure and proceeed to cool down the RCS to 425°F.

2. At 1000 psig and 425°F, the operator will close and lock out the accumulator isolation valves.

The above plant operating procedures will ensure that the accumulator isolation valves will not be locked out prior to about 2-1/2 hours after reactor shutdown for a cooldown rate of 50°F/hr.

A conservative analysis has defined that the peak clad temperature resulting from a large break LOCA would be significantly less than the 2200°F Acceptance Criteria limit using the ECCS equipment available 2-1/2 hours after reactor shutdown.

The following assumptions were used in the analysis:

1. The RCS fluid is isothermal at a temperature of 425°F and a pressure of 1000 psig.

2. The core and metal sensible heat above 425°F has been removed.

3. The hot spot occurs at the core midplane.

4. The peak fuel heat generation during full power operation of 12.88 kW/ft (102% of 12.63 kW/ft) will be used to calculate adiabatic heatup.

5. At 2-1/2 hours decay heat in conformance with Appendix K of 10 CFR 50, the peak heat generation rate is 0.179 kW/ft.

6. Two low head safety injection pumps and one high head charging pump are available from either manual Safety Injection actuation or automatic actuation by the containment Hi-1 signal.

7. No liquid water is present in the reactor vessel at the end of blowdown.

8. A large cold leg break is considered.

For a postulated LOCA at the cooldown condition of 1000 psig, previous calculations show that the clad does not heat up above its initial temperature during blowdown. Proceeding from the end of blowdown and assuming adiabatic heatup of the fuel and clad at the hot spot, an increase of 446°F was calculated during the lower plenum refill transient of 89 seconds. During reflood, the core and downcomer water levels rise together until steam generation in the core becomes sufficient to inhibit the reflooding rate. At that time, heat transfer from the clad at the hot spot to the steam boiloff and entrained water will commence. This heat removal process will continue as the water level in the core rises while the downcomer is being filled with safety injection water. The reflood transient was evaluated by considering two bounding cases:

1. Downcomer and core levels rise at the same rate. No cooling due to steam boiloff is considered at the hot spot. Quenching of the hot spot occurs when the core water level reaches the core midplane.

2. Core reflooding is delayed until the SI pumps have completely filled the downcomer. No cooling due to steam boiloff is considered at the hot spot until the downcomer is filled. The full downcomer situation may then be compared with the results of the ECCS analysis in the SAR to obtain a bounding clad temperature rise thereafter.

For Case 1 described above, the water level reached the core midplane 43.2 seconds after bottom of core recovery. The temperature rise during reflood at the hot spot from adiabatic heatup is 216°F, which results in a peak clad temperature of approximately 1086°F.

For Case 2, the delay due to downcomer filling is 54.4 sec. The corresponding temperature rise at the hot spot form adiabatic heatup is 272°F, which gives a hot spot clad temperature of 1143°F.

The clad temperatures at the time when the downcomer has filled for the DECLG, C_D = 0.6 submitted to satisfy 10 CFR 50.46 requirements are 1620°F and 1774°F at the 6.0 and 9.0 foot elevations, respectively.

Core flooding in the shutdown case under consideration will be more rapid from this point on due to less steam generation at the lower core power level in effect; decay heat input at any given elevation is less in the shutdown case. The combination of more rapid reflooding and lower power in the fuel insures that the clad temperature rise during reflood will be less for the shutdown case than for the design basis case.

Utilizing the preceding approach, the time calculated to just initiate an uncovery of the core is 13 minutes. The conclusion is that even for the conservative method outlined above, there exists adequate margin to retain a safe core condition even in relation to a ten minute operator-response-time assumption."

These operator requirements are verified, in general, by reference 12, SER Supplement 2 page 6.6-6.8 under "Emergency Core Cooling System - Performance Evaluation," and pages 7-1 and 7-2 under "Upper Head Injection Isolation Valves."

Additionally, the status of the ECCS systems from entry into the RHR MODE through cooldown, i.e., from 425 psig/350°F through MODE 5 is clarified by the following extract from reference 11, Suppl. SER No 1, pages 5-1 and 5-2 which confirms continuance of the alignment at the end of MODE 3 425 psig/350°F through both MODES 4 and 5.

"5.2.2 Overpressure Protection

In the Safety Evaluation Report we indicated a concern about the possibility of reactor vessel damage as a result of overpressurization when the reactor coolant system is water-solid during startup and shutdown. We have reviewed the applicant's system for overpressure protection when the reactor coolant system is water-solid. It consists of two separate trains each containing a power-operated relief valve set to open when the system pressure reaches 400 pounds per square inch gauge should an overpressure event occur. Each train contains an annunciator which sounds to alert the operator when plant conditions require enabling of the water-solid overpressure protection system; enabling is performed manually, by turning key-lock switch. The system is automatically disabled when plant conditions no longer require it; an annuciator sounds to indicate the system is no longer needed so that the operator may turn the key-lock to disable the system until needed. In addition, each train contains an annuciator which sounds when the power-operated relief valve is open, indicating an overpressure transient is in process.

Each power-operated relief valve is supplied with nitrogen from the cold leg accumulators. No operator action is required in the event of a transient. The operator isolates the upper head injection system, the cold leg accumulators, the safety injection pumps and one centrifugal charging pump before the reactor coolant system is cooled to 300 degrees Fahrenheit; only the remaining centrifugal charging pump could cause an overpressure transient as a result of inadvertent start with concomitant mass addition. The only other overpressure event would result from an inadvertent main coolant pump start with the coolant in the secondary side of the steam generator hotter than that in the reactor coolant system. The applicant has shown that in neither case was 10 CFR Part 50, Appendix G limit reached. For the latter case (that for main coolant pump inadvertent start), the applicant assumed that the temperature of the fluid in the steam generator would exceed that in the reactor coolant system by no greater than 50 degrees Fahrenheit.

The staff requires that the technical specifications require that the reactor coolant system may not be cooled to temperatures lower than 300 degrees Fahrenheit without the overpressure protection system enabled, and unless both and T.S. temperature constraints, would permit an RCS temp of 557°F. The only available analysis in the Licensing Basis, see earlier under "General," shows that cooling down to [1000 psig]/425°F is necessary to reduce the thermal burden on the ECCS so that the reduced ECCS capability can mitigate the consequences of a LOCA to 10 CFR 50.46 requirements; reference 8, pages Q 212-61, revision 28 and Q 212-61a, revision 28. The current T.S. is therefore non-conservative in this matter, and the licensee must evaluate and propose. Note; the "Footnote* Pressurizer Pressure above 1000 psig" also needs amendment.

Item: 3.5.1.1.d.

Nitrogen cover pressure is quoted at between 400 and 454 psig. The Licensing Basis FSAR, reference 4, page 1 of 5 revision 39 in Table 6.3.2-1 specifies a normal operating pressure of 427 psig. Making an allowance for channel error and drift should not this value be a higher set point of approx. 450 psig. The specified set point values proposed in the T.S. of 400 to 454 psig can therefore give actual values which are lower than in the Licensing Basis FSAR and be non-conservative. The Licensee shall evaluate and propose.

Item 3.5.1.1.f Proposed

The NRC proposes that an additional item limiting the range of actual water temperature in the accumulator between 60-150°F in accordance with Licensing Basis FSAR reference 29, Table 6.3.2-1 is necessary to confirm Safety Analysis Limits for this accumulator. Its absence from the proposed T.S. renders it potentially non-conservative. Further Item 4.5.1.1.1.a. concerning verification parameters should include Temperature of Accumulator Water. The licensee shall evaluate and propose.

ACTION Items a and b require HOT SHUTDOWN generally, except for closed isolation valves. This may be too conservative - the licensee should review specific cases identified under 3.5.1.1.a-f and decide whether HOT SHUTDOWN is necessary instead of to 1000 psig/425°F. Further, is there any conservative direction of the error which may minimize his need to suspend operations at power, or allow him to operate at reduced levels. This licensee proposal may be unecessarily conservative. The licensee may evaluate and propose.

Item 4.5.1.1.c requires that "once per 31 days when the RCS pressure is above 2000 psig, it is verified that power to the isolation valve on the Cold Leg Injection Accumulator is disconnected. What is the safety basis for this action, and where is it discussed in the Licensing Basis FSAR.

Item 4.5.1.1.1.d.1 requires that

"At least once per 18 months verify that each accumulator isolation valve opens automatically under each of the following conditions:

1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint,"

We are not aware that this actually occurs; the licensee shall review and advise of the related details within the FSAR on other licensing basis records. This action is not described in FSAR reference 7, under Table 7.3.1-3 (1 of 2)

Item 3.5.1.2.d: Proposed.

It is proposed that an additional item limiting the range of actual water temperatures in the accumulator to between 70 and 100°F in accordance with reference 29, Page (1 of 5), revision 39, in Table 6.3.2.1 is necessary to confirm the Safety Analysis Limits for the UHI Accumulator. It is also proposed that it be added as an additional surveillance element to item 4.5.1.2.a. Its absence from the proposed T.S. renders it potentially non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Action Items a & b require HOT STANDBY, generally, except for closed isolation valves, followed by HOT SHUTDOWN. This may be too conservative - the licensee should review specifically each of the Operability items b, c and proposed d, and decide whether HOT STANDBY leading ultimately to HOT SHUTDOWN is necessary. Further, he should assess if either boundary value, upper or lower, can be conservative, and by how much, and evaluate whether he should take an ACTION STATEMENT under "conservative" conditions. The licensee may evaluate and propose.

The licensee shall verify that the relief valve set point on the Accumulator is included in the In Service Testing Program at the facility.

T.S. Section 3/4.5.1.b (Proposed)

An additional T.S. item is proposed that provides specifically for the fact that "UPPER HEAD INJECTION SYSTEM ISOLATION VALVES" at APPLICABLE CONDITIONS of MODE 3 ($<$ 1900 psig and $>$ 425°F), MODE 4 and MODE 5, would have a "LIMITING CONDITION OF OPERATION" providing that "Each upper head injection system isolation valve" is closed and gagged. The UHI hydraulic pump and the gag motors for the UHI isolation values are de-energized and tagged. Appropriate Action Statements and Surveillance Procedures would be provided. This in accordance with the LCOs of the Licensing Basis FSAR as described in earlier items T.S. 3/4.5, "GENERAL" and T.S. 3/4.5.1 of this review.

Absence of this specific provision makes the current T.S. non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

T.S. Section 3/4.5.2 ECC SUBSYSTEMS -Tavg $\geq$ 350°F

The title should be amended to read as:

ECCS SUBSYSTEMS - PRESSURIZER PRESSURE $\geq$ 1000 psig/RCS Tavg$\geq$425°F

The Operability requirements of 2 full trains of ECCS equipment remains unchanged.

Absence of the pressure/temperature condition in the proposed T.S. is not in accordance with Safety Analysis Limits. Its absence permits high pressure pump operation at lower pressures and temperatures with potential infringement of related safety criteria. Related safety criteria have not been well defined, or docketed, but are apparently considerations of Low Temperature Overpressure Protection of the RCS under these and related Accident circumstances including inadvertent operation of ECCS pumps. This diversion from the Safety Analysis

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Item 3.5.1.2.d: Proposed.

It is proposed that an additional item limiting the range of actual water temperatures in the accumulator to between 70 and 100°F in accordance with reference 29, Page (1 of 5), revision 39, in Table 6.3.2.1 is necessary to confirm the Safety Analysis Limits for the UHI Accumulator. It is also proposed that it be added as an additional surveillance element to item 4.5.1.2.a. Its absence from the proposed T.S. renders it potentially non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Action Items a & b require HOT STANDBY, generally, except for closed isolation valves, followed by HOT SHUTDOWN. This may be too conservative - the licensee should review specifically each of the Operability items b, c and proposed d, and decide whether HOT STANDBY leading ultimately to HOT SHUTDOWN is necessary. Further, he should assess if either boundary value, upper or lower, can be conservative, and by how much, and evaluate whether he should take an ACTION STATEMENT under "conservative" conditions. The licensee may evaluate and propose.

The licensee shall verify that the relief valve set point on the Accumulator is included in the In Service Testing Program at the facility.

T.S. Section 3/4.5.1.b (Proposed)

An additional T.S. item is proposed that provides specifically for the fact that "UPPER HEAD INJECTION SYSTEM ISOLATION VALVES" at APPLICABLE CONDITIONS of MODE 3 ($\leq$ 1900 psig and > 425°F), MODE 4 and MODE 5, would have a "LIMITING CONDITION OF OPERATION" providing that "Each upper head injection system isolation valve" is closed and gagged. The UHI hydraulic pump and the gag motors for the UHI isolation values are de-energized and tagged. Appropriate Action Statements and Surveillance Procedures would be provided. This in accordance with the LCOs of the Licensing Basis FSAR as described in earlier items T.S. 3/4.5, "GENERAL" and T.S. 3/4.5.1 of this review.

Absence of this specific provision makes the current T.S. non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

<u>T.S. Section 3/4.5.2 ECC SUBSYSTEMS Tavg $\geq$ 350°F</u>

The title should be amended to read as:

ECCS SUBSYSTEMS - PRESSURIZER PRESSURE $\geq$ 1000 psig/RCS Tavg$\geq$425°F

The Operability requirements of 2 full trains of ECCS equipment remains unchanged.

Absence of the pressure/temperature condition in the proposed T.S. is not in accordance with Safety Analysis Limits. Its absence permits high pressure pump operation at lower pressures and temperatures with potential infringement of related safety criteria. Related safety criteria have not been well defined, or docketed, but are apparently considerations of Low Temperature Overpressure Protection of the RCS under these and related Accident circumstances including inadvertent operation of ECCS pumps. This diversion from the Safety Analysis

RCS pressures" of 2485 psig under these circumstances. Also the proposed T.S. alignment eliminates safety injection and charging pump capacity. There is no available evaluation of the capability of the reduced ECCS system to satisfactorily mitigate the consequences of a Small Break or Large Break LOCA from 2485 psig/350°F as is provided for the values of 425 psig/350°F within the Licensing Basis as described earlier under T.S. 3/4.5, Item: GENERAL. Our evaluation is that the absence of this pressure condition is non-conservative, and especially with respect to the Safety Analysis Limits of the Licensing Basis. The Licensee shall evaluate and propose.

The proposed limit at COLD SHUTDOWN MODE 5 is conditioned by the fact that Refueling is a condition of a vented vessel with Reactor Vessel Bolts untensioned, and non-ECCS alignments are proposed to deal with related events. Reference 8 pages Q212-56 revision 25 under the Titles of Case 1 and Case 2 and page Q 212-57, revision 25, under the Title of Case 3. Overpressure Protection also, which is a principal determinant of alignment, also ceases with untensioning the Reactor Vessel bolts for refueling.

The proposed T.S. under this Section requires a minimum of one only ECCS subsystem comprising

a. One Operable Centrifugal Charging Pump (CCP)

b. One Operable RHR Heat Exchanger

c. One Operable RHR Pump

d. An Operable Flow Path

There are no Safety Analyses or Evaluations of one only ECCS subsystem allowing for a single active failure in one only train. This proposition is therefore non-conservative with respect to the Licensing Basis FSAR. The Licensee shall evaluate and propose.

This T.S. does not disallow the additional CCP and 2 Safety Injection Pumps (SIPs) from 350°F down to 300°. This again is non-conservative with respect to the LCOs of the Licensing Basis FSAR which allows only one (1) CCP, and the remainder i.e., one (1) CCP and any other reciprocating charging pump and 2 SIPs are to be electrically isolated against inadvertent operation. This proposed T.S. is again non-conservative in respect of overpressure protection when compared with the current Licensing Basis. The licensee shall evaluate and propose.

The proposed T.S. allows one (1) CCP and one (1) SIP whenever the RCS temp is less than 300°F. The LCO of the Licensing Basis FSAR allows only one (1) CCP because of OVEPRESSURE PROTECTION; reference earlier information under earlier T.S. Section 3/4.5. Item: "General". The proposed T.S. is therefore non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

The LCOs of the Licensing Basis FSAR require the same operability of ECCS equipment as is required for TS 3/4 5.2A Proposed. So that in addition to:

additional provision be made in the RWST. The licensee may evaluate and propose.

T.S. Section 3/4.5.5 REFUELING WATER STORAGE TANK

Item: APPLICABLITY MODES 1, 2, 3, 4.

The current MODES 1, 2, 3 and 4 which includes an LCO for 372,100 gallons must be extended to MODE 5 and MODE 6 (limited) to meet the FSAR requirements in reference 8, pages Q 212-57 and 58, revision 25, item: Case 3: [when] The RCS is depressurized and vented with the air in the steam generator tubes, with the reactor vessel head on, and tensioned - and later with open relief paths between the head and the reactor vessel cavity and refueling canal. The single failure of an RHR/RCS Isolation valve is resolved by the expected Operability of the RWST providing 5 hours of injection flow. The recovery description also means that the RWST must be available in MODE 6 until the vessel head is removed and the refueling canal is filled to its specified level. It must also be available at termination of core alterations - in Mode 6, when drainage of the refueling canal commences until the Reactor Vessel Head is tensioned, when the RCS then moves into MODE 5. The proposed T.S. is non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Action Statement: The proposed ACTION should be modified [ ] as follows:

With the RWST Inoperable, restore the tank to OPERABLE status within 1 hour, or be in at least HOT STANDBY [and borated to a boron concentration which will give a shut down margin of 1% delta k/k at 200°F and a minimum of 2000 ppm] within [the next] 6 hours and in COLD SHUTDOWN within the following 30 hours.

The Licensing Basis FSAR requires Safety Injection of 2000 ppm Boron to mitigate the nuclear power consequences of any accidents which may initiate during this period; if the RWST is not available, then Boron Concentration in the RCS should be increased to the level required to mitigate any potential return of nuclear power. The proposed T.S. appears nonconservative.

The licensee shall evaluate and propose and in so doing he should evaluate each of the Operability requirements separately to determine if COLD SHUTDOWN is required for each INOPERABILITY REQUIREMENT, or whether alternate mitigating Actions are possible.

to be in cold shutdown in the event of failure, there of, we must consider the proposed T.S. non-conservative. The Licensee shall evaluate and propose.

T.S. Page 3/4 7-4: AUXILIARY FEEDWATER SYSTEMS

Item: APPLICABILITY MODES 1, 2 and 3 in the proposed T.S. should be expanded to MODES 4 and 5 in accordance with our review under Table 3.3-3 ESFAS INSTRUMENTATION, Items 7 a, b, c, d, e, and f. The conclusions from that review are: The proposed T.S. items are generally non-conservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Item 3.7.1.2.b. The licensee has deleted OPERABILITY requirements for the Steam-Turbine driven auxiliary feedwater pump at steam pressures of less than 900 psig. This is not in accord with current Accident Analyses and no justification has been provided: Reference 15, Recommendation GL-3, requires the Steam-Turbine AFW pump in the event of complete loss of AC power for a period of 2 hrs and beyond. This will require operability down to the lowest pressures for which the Turbine is provided as described in reference 22, Table 10.4.7-6 where the range of operating pressures provided for is from 110 psig to 1205 psig. This will also provide for operabilty down to and including MODES 4 (and availability from MODE 5) to cover licensing requirements discussed elsewhere under Table 3.3-3, ESFAS INSTRUMENTATION, Items 7a through f.

We note two principal features relating to the service conditions of the Turbine Driven Feedwater Pumps:

a. They are supplied with steam from two steam generators from main steam lines after the flow restriction orifices at outlets from the Steam Generators.

b. They would normally be expected to perform early in the transient and continue to function to design flow requirements throughout the Occurrence.

The licensee should explain how the proposed TS ensures that the Turbine Driven pump maintains its flow performance required by Accident Analysis when steam line pressures could drop substantially below the Steam Generator Pressures due to presence of the SG flow restrictions and until main steam isolation valves are isolated on steam line pressure of less than 565 psig (< provides for channel drift and errors).

The licensee shall evaluate the above comments and propose technical specifications which will ensure operability of the Turbine-Driven AFW Pump over the range of conditions expected from Design Basis Accident Analysis, and other less bounding events, down to and including MODE 4 as discussed in the Licensing Basis.

In his evaluation, the licensee should advise if Item 1e of Table 3.3-5 ESFAS INSTRUMENTATION, Steam Line-Pressure Low is derived from steam line sensors and after the SG orifices, or if it is taken from pressure sensors on the Steam Generator. The licensee should then advise what has been used in assessing Steam Generator Pressure Response and Turbine Driven AFW pump response in the

APPLICABILITY MODES proposed are 1, 2 and 3, with lesser volumes required in MODES 4 and 5.

ACTION STATEMENT should include a provision that, with the condensate storage tank inoperable, within 4 hours either

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

Demonstrate the OPERABILITY of the Nuclear Service Water System and Standby Nuclear Source Water Pond (alternate water source) as a backup supply, and align to the auxiliary feedwater pumps, and restore the condensate storage tank to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS should include

a. The condensate storage tank system shall be demonstrated OPERABLE at least once per 12 hours by appropriate measures when the tank is the supply source for the auxiliary feedwater pumps.

b. The Nuclear Service Water System and Standby Nuclear Source Water Pond shall be demonstrated OPERABLE at least once per 12 hours by appropriate measures.

Additionally, an evaluation of and provision will need to be made concerning potential loss of AFW supplies during loss of suction and change-over to alternate AFW sources.

The safety basis for these requirements are

a. Our earlier review under TS. Table 3.3-5 Items 7a and 7b show that whereas all safety evaluations involving AFW supply have assumed a Safety Analysis Limit of 61 sec. response time, this is only available from nonsafety related water sources. Further, that the safety related supply from the Nuclear Service Water Pond may take an extra 15 secs which is substantially non-conservative in respect of the related safety analysis.

Therefore, at this time, until the licensee has evaluated our concerns and made acceptable proposals, the NRC will require technical specifications on this non safety-related water storage of the above nature. The proposed T.S. are nonconservative with respect to Regulatory Requirements. The licensee shall evaluate and propose.

<u>T.S. Page 3/4 7-8: MAIN STEAM ISOLATION VALVES</u>

Item 3.7.1.4. The proposed T.S. provides that: "each main steam line isolation valve (MSLIV) shall be OPERABLE with APPLICABILITY MODES 1, 2, and 3.

to the SG PORVs is contrary to Regulatory Requirements which have been excluded from the Licensing Basis. The Licensee shall evaluate and propose.

T.S. Section 3/4.7.3: COMPONENT COOLING WATER SYSTEM

The proposed T.S. requires that:

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The SER for the plant under reference 10, summarizes the following Licensing Basis for the Component Cooling System:

9.2.4 Component Cooling System

The component cooling system provides cooling water to selected nuclear auxiliary components during normal plant operation and cooling water to safety-related systems during postulated accidents.

The component cooling system is designed to: (1) remove residual and sensible heat from the reactor coolant system via the residual heat removal system during shutdown; (2) cool the letdown flow to the chemical and volume control system during power operation; (3) cool the spent fuel pool water; and (4) provide cooling to dissipate waste heat from various primary station components during normal operation and postulated accident conditions. Active system components necessary for safe plant shutdown are designed to include at least 100 percent redundancy. The component cooling water for each unit includes two component cooling heat exchangers, four component cooling pumps and a split-volume component cooling surge tank. Two pumps and one heat exchanger per unit provide the necessary cooling water for normal operation, cooldown, refueling, and postulated accidents. The remaining pumps and heat exchangers serve as standby. An assured supply of makeup is provided from the nuclear service water system to each redundant loop.

The component cooling water system is designed to seismic Category I requirements, except for certain branches to non-essential equipment. The component cooling water pumps are powered by redundant emergency buses. The portion of the component cooling water system serving the residual heat removal system meets the single failure criterion for active components.

Based on our review, we conclude that the component cooling system design is in conformance with the requirements of General Design Criterion 44

The Licensing Basis FSAR, reference 6, page 9.2 - 12 , revision 39, item 39, provides for an allowable maximum of 94° which meets both maximum allowable temperatures for all Safety Related Components including NPSH requirements (reference 6, page 9.2-13, last para).

An average water temperature of 70°F has been selected by RSB as a potential design basis for Condition II, III and IV occurrences. The licensee has provided little information on the range of AFW temperatures used in his analyses and the related sensitivity of results to AFW temperature variations. In the Major Rupture of A Main Feedline, reference 7, page 15.4 - 13, it is stated that a "relatively cold (120°F) AFW temperature was used (after purging the feedwater lines)." "Excessive Heat Removal" analyses in reference 7, page 15.2 - 29, uses a "conservatively low feedwater temperature of 70°F."

We note that reference 6, page 9.2-13, revision 39, item 8 discusses ice formation on the surface of the pond which would imply near freezing temperatures for water supply. At this time, we have no record of any Safety Analysis being undertaken at such low inlet temperatures and on this basis we must consider any such low value as non-conservative.

The licensee will advise the range of AFW temperatures used in Condition II, III and IV events, their sensitivity to AFW temperature values, and from this his bases for setting any alternate values proposed to the water temperatures in the standby nuclear service water pond. The proposed TS maximum value of 78°F is conservative with respective to certain Accident Analyses; the lack of a minimum temperature of 70°F including possible near-freezing temperatures must be considered as nonconservative in respect of certain events. The Licensee shall evaluate and propose.

APPLICABLE MODES: The system is required in all MODES 1, 2, 3, 4, 5, & 6 to handle heat rejection requirements as the ultimate heat sink. The licensees proposal to limit this to MODES 1, 2, 3 and 4, is nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

Reference 6, page 9.2-13, revision 39, states that "In the event of solid layer of ice" forms on the SNSWP, the operating train [of the Nuclear Service Water [NSW] system] is manually aligned to the SNSWP. The Licensee shall provide the Safety Related reason for this action and advise if this operator action conflicts with the Response Times proposed under Table 3.3-5. Given a Safety Related reason, surveillance requirements ensuring this action should be included under either T.S. Section 3/4.7.5 NSWS or this particular T.S. Section 3/4.7.5 STANDBY NSWP. Absent this surveillance requirement on a Safety Related Issue, the proposed T.S. would be non-conservative. The Licensee shall evaluate and propose.

Why are T.S.s not applied to the closure of these valves also. The proposed T.S. may be nonconservative with respect to the Licensing Basis. The licensee shall evaluate and propose.

We also note an apparent non-conservative discrepancy between the basis for the specified reactivity condition of "a k_{eff} of 0.95 or less" without any specification of the position of movable control assemblies. We also note the need to add, according to reference 7, page 15.2-14, revision 10, that the boron concentration is to give a shutdown margin of at least 5 per cent delta k with all the rod cluster control assemblies out. The additional requirement underlined should be a part of the LCO for this T.S. item. Without this provision in the proposed T.S, it could be interpreted as non-conservative in respect of the Safety Analysis limits for the plant. The licensee shall evaluate and propose.

In the Licensing Basis FSAR reference 8, page Q 212-24, item 212.57, it is required that the reactor makeup water pumps shall be removed from the loads supplied by the emergency power supplies. This is to prevent inadvertent boron dilution during certain Occurrences in which electrical loads are disconnected from, and returned to, the Emergency Buses. Provision should be made so that at the end of refueling, before start-up, a surveillance procedure will confirm that this Licensing Basis FSAR requirement continues to be met. Absence of confirmation of this LCO is a non-conservative condition; the licensee shall evaluate and propose.

T.S. Item 3/4 9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION; HIGH WATER LEVEL

The LCO provides that:

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

The Licensing Basis, reference 20, Page 5.5-23, under Refueling, and page 5.5-24 under 5.5.7.3.1, System Availability and Reliability, last paragraph, shows the licensing of the RHR system is never based on only one RHR system being operable. Two are always to be available. This proposal is therefore outside the LCO for the FSAR in a non-conservative manner. The Licensee shall evaluate and propose

In his Basis, on T.S. Page 3/4 9-2, last para., the licensee has proposed that:

> "With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core."

In the FSAR, reference 8, page Q 212-56 under Case 2, it has been estimated that on loss of all RHR Cooling due to a fail closed RHR/RCS isolation valve, it will take 2½ hours for the available water inventory to boil. In that case, a number of alternates are proposed to resolve the situation and almost invariably, electric power is required, and in most cases the RHR equipment is used. If the basis for the licensee's request here is to enable him to operate

Review of available responses to the consequences of a fail closed RCR/RHR isolation valve, include many procedures using the containment sump. To allow for this single failure contingency, the licensee should therefore ensure that the containment sump will be operable during this mode, and with an appropriate surveillance procedure. There should also be provision for available fire pumps and necessary hoses to be assuredly available to enable use of the alternate procedures which have been described in reference 8, pages Q 212-56 and 57, revision 25. The current T.S. must be considered non-conservative. The licensee shall evaluate and propose.

T/S Page 3/4 9-12 REFUELING OPERATIONS -

The subtitle should read as 3/4.9.9 HIGH WATER LEVEL

Clarify by addition of the term HIGH

T/S Page 3/4 9-11 REFUELING OPERATIONS LOW WATER LEVEL

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is less than 23 feet.

GENERAL REVIEW: Whereas the existing FSAR under reference 20, page 5.1-7 discusses Refueling, it does not provide for a sustained period of normal operations under these Low Water Level conditions. The FSAR provides that:

> "Refueling
>
> Before removing the reactor vessel head for refueling, the system temperature has been reduced to 140°F or less and hydrogen and fission product levels have been reduced. The Reactor Coolant System is then drained until the water level is below the reactor vessel flange. The vessel head is then raised as the refueling canal is flooded. Upon completion of refueling, the system is refilled for startup."

Furthermore, we find that the FSAR analyses of the single failure of the RHR/RCS isolation valve is not predicated upon operations at "Low Water Level" so that no specific analyses and/or protective actions have not been developed for these circumstances. However analyses have been undertaken for the water inventories and temperatures in the RCS system that might apply under those conditions. Presumably therefore, the "OPERATING MODE - LOW LEVEL" is a long term changing condition following Cold Shutdown, with loops drained and bolts tensioned changing to bolts untensioned and removal of the head, as concomitant flooding of the reactor vessel cavity continues. At this time therefore, we cannot presume that the consequences of the case of single failure of the RHR/RCS isolation valve used as Case 3 in FSAR reference 8, page Q21-57, does not also apply under this MODE. We will use these consequences to evaluate.

Further, since this is effectively a long term changing condition, in the FSAR, it is not acceptable to allow some of the provisions requested such as one hour for the performance of CORE ALTERATIONS--which by T.S 3/4 9.9 are only permissible under that specification with at least 23 feet of water over the reactor vessel flange.

Footnote *: provides that,

"*Prior to initial criticality the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs."

This is an invalid request as all CORE ALTERATIONS are only permissible under TS 3/4 9.9 HIGH WATER LEVEL - REACTOR VESSEL. This is a non-conservative T.S proposal. The Licensee shall propose and evaluate.

Item 4.9.8.2, a surveillance requirement, specifies:

"At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours."

A time delay of 12 hours is excessive to verify a loop in operation, and this has been considered earlier in this section.

Further, the surveillance requirement, every 12 hours, is intended to ensure not only that the system is operating, but that it is operating at process conditions, including instrumentation and control, which can be evaluated to show that the equipment is capable of performing its Licensing Basis safety function. The current requirements for this item are absent most of this information; it is therefore non-conservative and the licensee shall evaluate and propose.

The current ACTION STATEMENT calls for containment closure in 4 hours [i.e. 240 mins]. Earlier conservative calculations for this MODE show that loss of all RHR in this MODE can cause boiling in 5 minutes and core uncovery in 100 mins. Given the circumstances, containment enclosure should be effected immediately, commencing RHR low flow alarms. The licensee shall evaluate, and propose. The current T.S. appears nonconservative with respect to the Licensing Basis.

Addenda

T.S. SECTION 3/4.5 EMERGENCY CORE COOLING SYSTEMS

T.S. SECTION 3/4.4.4.1 RCS LOOPS AND COOLANT CIRCULATION/HOT SHUTDOWN MODE 4

More recent information, and a detailed check on certain elements of the proposed T.S. relevant to the above section, and the Licensing Basis FSAR, and particularly reference 5, Section 7.4.1.6 Emergency Core Cooling Systems and Section 7.4.1.5 Residual Heat Removal System, does not appear to prov[illegible] acceptable surety that:

a) The Reactor Coolant Pressure Boundary (RCPB) valves on the RHR/RCS suction line are confirmed closed in MODES 1, 2, & 3.

b) That the RCPB valves in the RHR/RCS suction line are individually identified as opened in the RHR MODE.

c) That in RHR MODE 4, the RHR system must be capable of automatic re-alignment to the ECCS mode with residual ECCS equipment, in the event of a SI signal, including automatic closure of the RCPB Isolation valves on the RHR/RCS Suction Line in accordance with 10 CFR 50 App A Criterion 55(4) and subsequent automatic opening of valves to the RWST in accordance with 10 CFR 50 App A, Criterion 20 [with appropriate provision for RHR pump protection].

The current position in respect of c above appears to be absent those requirements and therefore non-conservative. The Licensee shall evaluate and propose.

The T.S. should provide the LCOs and surveillance in the overpressurization protection system of the RHR system as described in Licensing Basis FSAR, reference 3, page 5-5-24.

Proposed T/S Page 3/4 5-6, item 4.5.2.d, 1) b) appears incorrect: it provides that, in establishing ECCS operability:

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR System from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 560 psig the interlocks will cause the valves to automatically close.

Item b) above is incorrect in that it should ensure that with a simulated or actual Reactor Coolant System pressure signal greater than 475 psig, the interlocks will cause [illegible] valves to automatically close, reference 4, section 5.5.7.3.3 and reference 5, section 7.4.1.5.4.

The proposed T.S. closes the valves when they are in fact required to be open and is therefore non-conservative. Further, the lower pressure of 475 psig required to close is more conservative than a valve of 560 unless there are Set Point and Channel considerations - The pressure is less conservative than the Licensing Basis FSAR value.